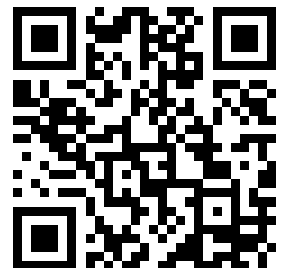

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Seminar on

INTENSE NEUTRON SOURCES

Proceedings of a United States Atomic Energy Commission/
European Nuclear Energy Agency Seminar

Santa Fe, New Mexico, 19–23 September 1966

Jointly organised by the
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(Local host : Los Alamos Scientific Laboratories)

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F O R E W O R D

Neutrons have been used as a research tool in the physical sciences for over twenty years. As in many technical fields, progress is accompanied by increasing sophistication and complexity of systems and inevitably leads to specialization and higher costs. Thus the seemingly simple questions, "What is to be done and how can it best be accomplished?", become more and more difficult to answer. Research physicists will of course continue to pose new problems, develop new techniques and request more varied and intense sources; design engineers, on the other hand, will continue to propose and develop more advanced and varied sources. One would hope that these two disciplines would join naturally to attain common goals and, indeed, this does occur both within laboratories and often between different organizations. It is most important to foster such cross-fertilization of ideas between related, but distinct, disciplines on as broad a foundation as possible, and international meetings serve this purpose.

When the European American Nuclear Data Committee and European American Committee on Reactor Physics proposed a Seminar on Intense Neutron Sources for research, it was clear that no common solution or approach, applicable to all needs, would be found. Nevertheless, the technical papers and discussions in these Proceedings contain the elements of the research problems of major interest today, along with the prospects for source designs to support and carry out this work.

Several trends seem to have emerged from the Seminar:

- recently designed and constructed steady state fission reactors are approaching the apparent limits of technology and economy;
- a healthy competition exists between advocates of continuous and pulsed fission systems, both from the research and design aspects;
- accelerators often exhibit a unique and sometimes complementary nature both as continuous and pulsed sources, and with and without fissionable-material "boosters";
- there is an increasing interest in solid state studies - both in the investigation of the static (structure) and dynamic

(vibrational states, etc.) characteristics of condensed matter.

The Editors have attempted to retain all of the technical arguments presented at the Seminar but, in the interest of conciseness, admit to editing of the discussions, often without consulting the speaker. Only in the case of the four Panels did contributors have a guaranteed opportunity to edit their own statements. The Scientific Secretaries of each Session and Panel contributed greatly to the preparation of these Proceedings. These Secretaries and the Editors are Staff Members of the Los Alamos Scientific Laboratory where the master manuscript was prepared.

Thanks are due to authors of the technical papers, to those who contributed to the discussions and panels, to the Scientific Secretaries and Session Chairmen, and to the local Organizing Committee who worked so diligently to make this a pleasant and profitable meeting. Lastly, thanks are due to the sponsoring agencies, the United States Atomic Energy Commission and the European Nuclear Energy Agency, and to the Los Alamos Scientific Laboratory for their generous support and encouragement.

H.T. Motz and G.R. Keepin
Co-editors

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SESSION I

BASES FOR INTEREST IN HIGHER FLUX

STUDIES OF CONDENSED MATTER USING NEUTRONS†

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ABSTRACT: Certain properties of condensed matter can be investigated with neutrons. Outlined in this paper are the advantages that neutrons have for these measurements and the success that has been achieved in (a) locating atoms in condensed matter, (b) observing the microscopic motion of the atoms and (c) determining the effects of neutron damage. Speculations are made of the future applications of neutrons, especially when higher intensity sources become available.

† Work performed under the auspices of the U. S. Atomic Energy Commission

1. INTRODUCTION

The investigation of the physical properties of condensed matter with neutrons⁽¹⁾ has developed rapidly even though neutrons are relative newcomers to this mature field of science. Neutrons were discovered in 1932 and the feasibility of using them in solid state research was quickly demonstrated. The experiments started to proliferate when intense sources of neutrons became available after the first reactor was developed in 1943. After this time the development of instruments to be used with reactors was forthcoming while more intense and new types of neutron sources became available. Simultaneously the study of the structure of matter, the dynamical properties of matter, and damage induced by neutron radiation advanced rapidly. An outline showing the highlights of this development is given in Figure 1 which also shows the advantages that studies using electromagnetic radiation have because of their earlier application.

Structure studies, the determination of the locations of atoms in crystals, was the first phase of solid state physics to be investigated by neutrons⁽²⁾. Once it was discovered that the neutron has a wavelength and intense enough sources of neutrons became available, the time was ripe to develop this field of physics. It was shown quickly that neutrons are better than electromagnetic radiation for studying certain properties of condensed matter, such as the locations of light atoms in the presence of heavy atoms and the determination of magnetic and antiferromagnetic ordering. Structure studies have now advanced to the threshold of studying large organic molecules by neutron diffraction.

A study of the dynamical properties of condensed matter⁽³⁾, the movement of atoms within the materials, was delayed for several years because of the necessity of having even more intense sources of neutrons than the first reactors delivered. But in the early 1950's it was shown that the dynamical properties of materials could be studied with the neutron sources that were then available. From this time the study of these properties has advanced

rapidly and many new types of measurements have been made. The study of dispersion relations in crystals has been the most rewarding while the study of the vibrational spectra of molecules is beginning to reveal spectra unseen by infrared and Raman methods.

The effects of radiation damage⁽⁴⁾ have been pursued experimentally more on the macroscopic level to observe neutron damage as it is revealed in the metallurgical and engineering properties of matter while these effects are explained on a microscopic level. These experiments have been restricted from using precisely tailored neutron fluxes, such as beams of monochromatic neutrons free of γ rays, because of the high dosages of radiation required to induce measurable effects. This radiation damage field can profit from the development of more microscopic techniques to observe the damage and by the use of tailored beams to better define the interactions. Both of these may be supplied by more intense neutron sources.

One sees from this chronological outline that X-ray crystallography started 40 years before neutron crystallography started and that infrared spectra was used 30 years before the neutron spectra measurements were developed. Thus the neutron, being such a relative newcomer, is best used to fill gaps in the X-ray crystallographic and infrared spectra measurements where the unique properties of the neutron can reveal properties of the physical states of matter that were not revealed by these earlier methods.

In the radiation damage field, intense tailored beams of protons, electrons, heavy charged particles and X-rays are now available while such beams of neutrons are still in the future. The unique properties of neutrons guarantees the use of such beams when they are available. If we allow the same lead time as with crystallography and spectra, intense neutron sources which can produce these beams should become available in the late 1970's.

This paper will outline briefly where the investigation of the physical properties of condensed matter with neutrons now stands and where it may hope to go if more intense sources of neutrons are made available. It appears

obvious that neutrons will continue to play as important a part, or a more important part, in the study of condensed matter as they do now whether more intense sources of neutrons are made available or not. The crystallographic studies of magnetic ordering and the locations of hydrogens in organics will continue, dispersion relations for many hundreds of crystals will be measured and the radiation damage studies will become more detailed.

II. STRUCTURE STUDIES

Consider first the section of solid state physics in which neutrons determine the locations of atoms in crystals. Neutrons have several distinct advantages for these measurements, such as (a) the study of the positions of light atoms in the presence of heavy atoms, (b) the location of atoms in the presence of neighboring Z atoms, and (c) the determination of the magnetic ordering of atoms in crystals.

Such structure determinations of materials using diffracted neutrons have advanced to the point where the measurements are as precise as X-ray diffraction measurements but they require high flux reactors as sources of neutrons and large spectrometers to obtain the data. These spectrometers are highly programmed and the data is analyzed through a large computer to give precise representations of the locations of the atoms and the average motion of the atoms. This programming has advanced to the point where stereoscopic pictures are plotted automatically by the computer. Figure 2 is an example of a recently determined structure of ammonium sulfate and shows the stereoscopic picture⁽⁵⁾ drawn by the computer. The small ellipses around each atomic location indicates the average motion of the atoms. Neutron crystallographic measurements are now being conducted at many labs around the world measuring more and more complex crystallographic structures. The field has advanced to the point where partial patterns are being obtained for neutrons⁽⁶⁾ scattered from proteins to supplement the measurements obtained by X-rays.

For neutron crystallographic studies, the instruments were developed

in the late 1940's and the initial measurements were made at that time. Since that time the instruments have been improved and refined for specific applications but the techniques have followed classical X-ray methods. Recently a uniquely different time-of-flight technique was developed and tried at several labs^(7,8,9). The promise of this method with pulsed sources of neutrons has been shown and extensions of crystallography using high intensity pulsed sources show great promise. For example, the time-of-flight diffractometer at the IHR at Dubna will have in the near future a resolution of $\Delta\lambda/\lambda = 0.5\%$ for all wavelength λ .

As an illustration of one of these time-of-flight instruments, Figure 3 shows a cutaway drawing of the MTR time-of-flight neutron diffractometer⁽⁸⁾. Here a beam of neutrons is taken from the reactor and a chopper chops this beam of neutrons into bursts of polychromatic neutrons. The neutrons which satisfy the Bragg condition are preferentially scattered from the sample into counters at 90° and 30° , and the neutrons wavelengths are sorted in time-of-flight over the two meter flight paths. In this particular application, the method is applied to scattering of neutrons from powdered samples which are under pressure. With the two 60-ton rams located at the sample position, pressures up to 40 kbars can be exerted on the sample. Thus the neutrons are employed to observe changes in phase or changes in magnetic ordering as a function of pressure. The advantage of this time-of-flight system with pressurized samples is that fixed angles are used which more nearly match the pressurized sample holders. In these holders it is necessary to bring the beam in through a small aperture and only have a small aperture through which the beam can exit.

Figure 4 shows an example of several time-of-flight spectra obtained with this machine when the sample was powdered $\alpha\text{-Fe}_2\text{O}_3$ ⁽¹⁰⁾. Here diffraction peaks from the neutron scattering from the magnetic ordering are observed and the change in the intensity of these peaks indicates that the magnetic ordering is being re-aligned as **temperature decreases**.

Each time-of-flight spectrum can be obtained in one hour's time even though the source flux is not the highest available. These data demonstrate the good resolution and counting rates of the time-of-flight method and the usefulness of neutrons to study samples under pressures.

The application of neutron diffraction techniques to the studies of liquids and molecular gases⁽²⁾ has not been as extensive as for solids. This is probably partially due to the lower symmetry of the liquid and gaseous states which make the interpretation of the data less definitive. Some of the difficulty in these experiments and their future are presented in the section on dynamical properties.

Where may the field of neutron diffraction be expected to go and what new information may be obtained? We can be sure that the study of crystallographic structures, particularly the locations of hydrogens, the ordering in alloys and of magnetic ordering will continue to expand as it has done for the last 15 years. More difficult structures will be analyzed, smaller samples will be used, and more precise measurements will be made. Still one will probably not use neutrons because of their limited availability for these measurements if X-rays which are plentiful can do the job. In cooperation with X-ray methods, neutron crystallography will be applied more to the complex organic molecules, such as the proteins, to not only locate the carbons and heavier atoms in the structure but to also locate the hydrogens and explain the hydrogen bonding which is so important in the organic molecules. The use of neutrons to study samples under high pressures has shown great promise and this field will expand up to pressures of 100 or more kbars. Here neutrons have a particular advantage because neutrons, as compared to X-rays or electrons, are a penetrating probe; not only can they penetrate through the sample containing material but they penetrate into the sample to see the sample in depth.

The time-of-flight system which has been shown to be competitive with the fixed wavelength diffraction methods will be used more extensively. If

this method is coupled to a pulsed source that yields 100 times the usable flux of existing sources, a great number of additional experiments will be possible. Single crystal measurements with time-of-flight which have only been surveyed so far will be perfected. Polarized neutron experiments will be possible by filtering the beams through dynamically polarized crystals⁽¹¹⁾ which produce well polarized beams.

While the time-of-flight method of neutron diffraction has developed with the anticipation of its use with intense pulsed sources, it may have equal impact on the research potential of low flux reactors such as are now available at several universities. By properly arranging many counters so that they cover the Debye-Sherrer cone and also so that they are on the curve of constant flight time for scattering from one plane of a powder, the intensity of scattering can be greatly increased. The curve for which the flight time is a constant is $D(\theta) = \text{const} [d_{(hkl)}/t_{(hkl)}] [1/\sin \frac{\theta}{2}]$ where $t_{(hkl)}$ is the flight time for the (hkl) planes as observed by the timing device, $d_{(hkl)}$ is the spacing of the (hkl) planes, $D(\theta)$ is the flight path over which the flight time is being measured and which is a function of the scattering angle θ . Counters covering greater than π solid angle are possible under these conditions.

Another modification of the neutron crystallographic experiment, one which recently has been applied with X-ray crystallography⁽¹²⁾, is the study of pulsed samples. Here the neutrons will be used to study the crystallographic orientation as different types of pulses such as magnetic or shock waves are applied to the sample. Samples that are repetitively pulsed in phase with the time-of-flight diffractometers would be ideally suited for such measurements. The pulsed reactors are on the verge of being intense enough sources to show complete diffraction patterns in one burst⁽⁷⁾ while the nuclear explosions are more than intense enough to show a pattern in each burst.

Of these future developments many will continue whether more intense sources become available or not, but progress will be slower if source

strength is not increased. Several experiments definitely require more intense sources. The determination of the locations of hydrogens in molecules as large as proteins has only been demonstrated, and factors of X 100 or more gain is needed to shift the emphasis of the measurements from development of methods to biophysical studies. The use of pulsed samples to study the transient phenomena requires more intensity.

III. DYNAMICAL PROPERTIES

Neutrons have certain advantages over other types of measurements for the investigation of the dynamical properties of condensed matter. One advantage is that both the wavelength of thermal neutrons and the energy of thermal neutrons are of the right magnitude to see both the momentum change and the energy change of scattered neutrons simultaneously. Thus the energy change and the coupled momentum change occurring in the sample can be observed simultaneously and the correlated effects are revealed. This is not true with either electrons or electromagnetic radiation. The second advantage of neutrons is that compared to electromagnetic radiation, they have a different set of selection rules for exciting transitions in molecules. This allows neutrons to observe excitations that are forbidden to infrared and Raman scattering. The third advantage is that since neutrons have a magnetic moment, magnetic structure and dynamical changes of this structure can be observed; thus neutrons can observe spin waves in crystals. The fourth advantage of neutrons is that they are penetrating particles; they do not see the surface of the sample but they see the sample in depth. Also the neutrons have a different power of interaction with the material as compared to electrons and electromagnetic radiation so that different particles in the sample can be observed with neutrons as compared to electrons and electromagnetic radiation.

These neutron advantages provide the best means of studying certain physical properties of condensed matter. Measurements of dispersion relations of the vibrational waves traveling through crystals have been

the most interesting and profitable solid state physical property to study with inelastically scattered neutrons. About 50 sets of dispersion relations⁽¹³⁾ have been measured. These measurements are characterized by the simpler structures and by those samples which are available in large single crystal sizes. These data are challenging the abilities of the theoretical physicists. Figure 5 shows an example of one set of dispersion relations⁽¹⁴⁾ for a sample of NaCl; both the acoustical branches and parts of the optical branches are shown. Recent measurements for aluminum⁽¹⁵⁾ indicate that the dispersion relations can be measured to an accuracy of less than 0.5%.

Besides the general physics of mechanical acoustical and optical dispersion curves, other special physical properties are being revealed. Dispersion relations for the magnetically coupled spin waves⁽¹⁶⁾ traveling through crystals have been measured for several samples. The interaction of the Fermi surface with the dispersion relations, the Kohn anomaly, has been observed⁽¹⁷⁾. The widths of the dispersion relations⁽¹⁸⁾ have been measured for a few samples which gives information about the anharmonic potential with which the atoms interact in the crystal. In one case the intensities⁽¹⁹⁾ of the neutron scattering peaks which define dispersion relations have been measured. These can be compared to structure factor calculations to give further evidence about the validity of the force model. Several samples have been varied in temperature⁽¹³⁾ to show the temperature effect of the dispersion relations and two measurements of the pressure effect^(20,21) on the dispersion relation have been reported. The effect of impurity atoms⁽²²⁾ on the shape of the dispersion relations and the width of these relations have been demonstrated. Within the last year a measurement of the magnon-phonon interaction⁽²³⁾ as revealed in dispersion relations has been presented.

Dispersion relations are obtained when neutrons are coherently scattered from single crystals. If the neutrons are incoherently

scattered from crystals, information about the frequency distribution of the vibrational waves in the crystal is obtained. However, nature has not provided many incoherent samples, and only a few attempts to measure frequency distribution by neutron scattering have been made⁽¹³⁾. Figure 6 shows an example⁽²⁴⁾ of the dispersion relations as measured for a sample of vanadium, the only sufficiently incoherent, naturally occurring sample. One notices that all of the curves have the same general shape but the details of the curves are quite different. It appears necessary to have an even more precise measurement of this frequency distribution before a detailed model for the forces in vanadium will be generated.

Neutrons have been used to reveal the dynamical properties of liquids such as the onset of diffusive motion in a liquid⁽²⁵⁾. The broadening of the quasi-elastically scattered peak of neutrons scattered from a liquid are related on a microscopic scale to the way the particles move away from their initial locations in the liquid. Figure 7 shows an example of this diffusive motion⁽²⁶⁾ for very short times. One notices that for times of less than 10^{-13} sec the atoms start to move away from their initial locations as if they were in a gas, that is with no force inhibiting their motion. At times between 10^{-13} sec and 10^{-12} the atoms are pushed back toward their centers or their initial positions, as if they were bound in a crystal. For longer times than 10^{-12} sec the atoms behave as if they were simply diffusing.

Recently dispersion relations or collective oscillatory motion for liquids have been measured as is demonstrated in Figure 8⁽²⁷⁾. Here the dispersion relations for a sample of liquid lead show not only the dispersion relations for longitudinal waves in the first Brillouin zone, but also show that this dispersion relation continues out through the second Brillouin zone. At present it is not clear how these dispersion relations define the forces in the liquid and more measurements and theoretical interpretation

of this type are needed.

Neutrons have been scattered from molecules in the solid and liquid phase to show intramolecular structure of the molecules⁽²⁸⁾. These experiments have usually been performed on predominantly incoherent scattering samples and have assumed that coherent scattering and intermolecular forces do not effect the observation of the intramolecular force transitions. Such measurements have confirmed transitions previously observed with infrared or Raman scattering but little new information has been revealed because of the difficulty of interpreting the experimental results. More precise measurements of the absolute intensities of the scattering at values of the momentum change where coherent effects are negligible are needed. These must then be interpreted by the fitting of curves derived from accurate models.

Recently the frequency distribution for the scattering of neutrons from stretch oriented polyethylene has been obtained^(29,30). While the molecules were not completely oriented, the semi-orientation did introduce enough symmetry into the experiment that correlation could be made between the peaks observed in the data and calculated peaks⁽³¹⁾ in the frequency distribution of polyethylene. These experiments show the necessity in the spectral measurements of limiting experimentally the number of variables and coupling the interpretation with detailed theoretical calculations.

Recently experiments on the scattering of neutrons from gaseous molecules⁽³²⁾ have assured that the intermolecular forces are not effecting the intramolecular transitions. Figure 9 shows the scattering of neutrons from molecules of ethane. The large elastic peak indicates that most of the neutrons were scattered with no energy change, while the peaks to the left and right show the excitation and de-excitation of the hindered rotational state in ethane, a transition that is forbidden to infrared and Raman scattering. It was necessary to go to small momentum changes because it is only at these small momentum changes that the peaks are

distinguishable from other effects in the scattering. Very small scattering angles of 4.2° and higher incident energies of 100 meV were required to guarantee that the momentum change observed in these experiments was small.

Recently neutrons have been scattered from deuterated methane gas⁽³³⁾ and a scattering law has been obtained as shown in Figure 10. Here one sees definite wave vector dependent structure related to the positions of the hydrogens with respect to other hydrogens and with respect to the carbon atoms. One not only sees this structure in the elastic scattering data but also in the inelastic scattering data. From these data it was concluded that it should be possible to measure molecular structure as accurately with neutrons as has previously been done with electrons but it will be necessary to not only observe the elastic scattering but the inelastic scattering also. It is the large inelastic scattering contribution and the structure in this inelastic scattering that demonstrates the need for dynamic information instead of diffraction data for a precise determination of the structure. A similar situation exists for liquids as demonstrated for liquid lead⁽²⁷⁾.

Neutrons observe the positions of the nuclei in the molecule while electrons and X-rays observe the electron cloud around the nuclei. The coupling of such data as given in Figure 10 with corresponding data obtained with electrons and X-rays provides a basis to compare differences in the shape of the electron cloud as compared to the positions of the nuclei. These should be fruitful measurements both for liquids⁽³⁴⁾ and molecules.

What future developments might be expected from neutron studies of the dynamical properties of materials? It is obvious that the measurements of the dispersion relations will progress and expand to become as broad a field as crystallography. While less than 50 dispersion relations have been measured so far, many thousands will be measured in the future.

The measurements will be extended to materials that have more atoms per unit cell which greatly complicates the dispersion relations because of the additional more-closely-spaced branches. Much smaller crystals will be used as samples, as small as one millimeter square. Samples with more absorption will be attempted which will require more intense neutron sources to observe the scattering. The samples will be varied in different ways, such as oriented samples. An example of this might be a vanadium single-crystal sample, held at close to 0°K in a strong magnetic field. Thus the vanadium atoms would be oriented⁽³⁵⁾, partially destroying the incoherent scattering condition and allowing the dispersion relations to be measured. This would be an important set of data to compare with the measured frequency distributions. More measurements of spin waves and magnon-phonon interactions will be made. The width of the scattering neutron lines leading to dispersion relations and the intensity of these lines will be measured in detail to reveal more about the anharmonic forces in crystals. Temperature and pressure will also be applied to the crystals to reveal more about the forces within crystals. Increased intensities and resolutions are desirable and perhaps necessary for these extensions.

For the measurement of frequency distributions, a breakthrough in interpretation or accuracy is needed. This may come with more careful measurements or with some manipulation of the sample which allows some of the parameters to be specified. It may be possible to measure frequency distributions by eliminating coherent scattering effects by using polarized beams in which only neutrons that are scattered with a change in polarization are observed⁽³⁶⁾. This change in polarization guarantees that the neutrons have incoherently scattered from the sample and the scattered spectrum is related to a frequency distribution. This frequency distribution measurement with polarized beams is similar to the frequency distribution measurement of the incoherent scattering from a single crystal⁽³⁷⁾. These measurements of frequency distributions will need to be precise and the effect of

multiple scattering and multi-phonon scattering must be carefully removed from the data. To measure frequency distributions from an incoherent sample prepared by mixing isotopes, in which case the sample will probably be small, increased intensity will be necessary to observe accurately the frequency distribution.

The future for measurements in liquids is not as predictable as in solids because the liquid state is less well defined at the present time and it is not as well known what should be measured. The measurements of dispersion relations in liquids looks promising and more extensive and accurate ones should be obtained. The short time diffusive motion has been demonstrated but more detailed information is needed to precisely define what is happening. Both of these type measurements, if they are successful, will be applied to more complicated samples.

It has been predicted that the Kohn anomaly⁽³⁸⁾ may be observed in the scattering from liquids. If it is observed, these measurements will lead to a detailed definition of the Fermi surface in liquids. The effect of the Kohn anomaly on the dispersion relations in liquids will be very small and careful measurements will be necessary.

At the critical point between liquid and gas, between antiferromagnetic and paramagnetic states, and between ferromagnetic and paramagnetic states long range collective oscillations are present. The exploration of this region with neutrons has been attempted and shows great promise^(16,39), but more intensity is required as careful measurements are needed. The coupled study of the diffraction pattern for a liquid as obtained by X-rays and the scattering law obtained from a liquid with neutrons should lead us to a better definition of the positions of the nuclei in a liquid and the corresponding position of the electron cloud around the nuclei. Some work has been done on this⁽³⁴⁾ by comparing the X-ray diffraction pattern to the neutron diffraction pattern. However inelastic scattering measurements on liquids and the measurement of the scattering law show that scattering law

is a much more sensitive and detailed picture of the positions and motions of the liquid than is a diffraction pattern. The coupling needs to be made with scattering law and not the diffraction pattern.

In the study of molecules, the experimental and the theoretical developments need to go along hand in hand for it is from the interpretation of the data with a detailed model that the most success has been realized. Experimentally more precise measurements need to be made with better resolution and with careful consideration for the effects of coherent scattering and how coherent scattering appears as a function of momentum change. The effect of multiple scattering must also be recognized and correction applied. Studies of the intramolecular transitions of molecules in gases are very promising at this time but the experiments have to be made at very small momentum change. This requires small scattering angles and higher incident energies where the intensities are small and the resolution is not good. Even more intensity is needed to get to the region of very small momentum change where precise measurements can be made on more complicated molecules. These measurements will supplement the infrared and Raman measurements revealing the transitions that are forbidden to these types of electromagnetic radiation.

In future neutron scattering experiments, discrete rotational levels in molecules and hindered rotational levels in molecules will be studied. It has been pointed out that vibration-rotation interactions have a pronounced effect⁽⁴⁰⁾ on the rotational level spacing, thus the pure rotational spectra are needed and will be observed with neutrons. Dispersion relations for large molecules will be obtained as is being attempted for stretch oriented deuterated polyethylene⁽⁴¹⁾. This can add a great deal of information about the intramolecular forces in these large molecules. As with liquids, the comparison between the scattering law for the coherent scattering of neutrons from molecular gases and the diffraction patterns of electrons and X-rays from these gases will be compared to show the relation between the electron cloud and the position of the nuclei in

the molecules.

Many of these future developments will continue whether a more intense source of neutrons is made available or not, but all of them will increase much more rapidly if more intense sources are made available. Certain measurements obviously need greater intensities. The study of dispersion relations on very small crystals and with particular sample manipulation require more intense sources. The accurate measurement of the intensity of the elastic scattering peaks requires more intensity as does the study of samples under pressure. The accurate measurement of frequency distributions and the measurement of frequency distributions with polarized beams and oriented samples require more intensity.

In the study of the dynamical properties of liquids, if Kohn anomalies are detected, more intense sources will be needed to extend the method to the determination of Fermi surfaces for many liquid metals. Since the studies of physical properties around the critical points are not well defined yet, these experiments will require more intensity. The study of intramolecular forces in molecules at very small momentum changes where the transitions can be measured precisely requires more intensity because of the lower scattering cross section and the need to improve the resolution.

IV. RADIATION DAMAGE

Neutrons have certain advantages for studies of radiation damage, both as a source of damage and as a diagnostic probe. As a generator of damage, the neutron is a non-ionizing particle and its primary interaction is with a single nucleus. This interaction can be deep within the sample since the neutron is a penetrating particle and it can either produce a dislocation, a transmutation, or a fission. As a diagnostic probe, some of the methods of Sections II and III are possibilities.

Radiation damage produced *by neutrons* at present employs integrated fluxes of either thermal or fast *neutrons* which produce dislocations or transmutations. The effects *of these* are observed in the change of the macroscopic properties such as elongation of a single crystal in particular crystal directions, a change in resistivity, yield strength, embrittlement, optical properties, etc. The observation of damage on a microscopic scale is more restricted because of the scarcity of sensitive methods. The dose rates⁽⁴²⁾ required for experiments are greater than 5×10^{12} neutrons/cm²sec to produce a significant effect in a reasonable time. To limit mobility of the defects and dislocations, some exposures are conducted with the samples cooled to as low as 4°K with the sample near a reactor core. The gamma heating in the sample and the coolant limits the rate at which the doses can be applied.

What advances might be expected in the fields of radiation damage? Although at the present time there appears to be relative contentment with the quality of the neutron fluxes, it seems possible that increased knowledge could be gained by better defining the neutrons producing the damage. Monoenergetic neutron fluxes or beams of monoenergetic neutrons would allow the energy threshold for damage to be investigated. Since about 25 eV is needed to produce a dislocation in a material, the energy range of most interest is the eV to KeV range. This, of course, is the range for which monoenergetic beams are most difficult to produce.

The availability of intense enough beams so that the sample under irradiation could be moved from near the core of a reactor to outside the shielding would allow greater variety in the experiments. The beams could be monoenergetic, and they could be free of gamma rays. Beams of neutrons permit the experiments to be performed with oriented crystals in which focusing and channeling effects can be studied. Such beams would allow for easy cooling and allow instrumentation of the sample for on-line or prompt observation of the effects. Using cold neutrons

which have energies below the Bragg cutoff for a material would allow the scattering cross section to be small so that the interaction would be mainly by absorption. In another type of experiment, by changing the incident energy of the beam from an energy corresponding to a strong scattering level to that for a strong absorption level would allow comparison of damage from dislocations and transmutations at nearly the same neutron energy.

At the present time cold samples are used to prevent annealing for a long enough time to allow the damage effects to be observed. With pulsed sources, one might observe the damage in such a short time that the temperature would not have to be lowered. With pulsed sources the rate of recombination might be observed as a function of temperature.

As indicated in Section II and III, neutrons as a diagnostic tool to observe impurities or dislocations are not sensitive as yet. To observe the effects of impurities in the frequency distribution and the dispersion relations required 1-5% admixture of impurity atoms. This represents $\sim 10^{22}$ atoms per cc which is past the limit of damage experiments. When more intense sources of neutrons become available these experiments will be possible with smaller impurity contents. In addition, very small angle scattering of neutrons can be employed to observe defect clusters. However, more intensity is needed to make these experiments precise.

V. CONCLUSIONS

Although the neutron is a relative newcomer for diagnosing the physical properties of matter in the condensed state, many methods for its use have been developed. Many of these are now as accurate as their predecessors and some show physical properties that are unobtainable by other methods.

Neutron diffraction measurements started immediately after the first

reactor neutron sources were obtained and are now as accurate as X-ray diffraction measurements. Future developments will be toward more complex systems with greater variety of manipulation of the samples, such as pressurizing and magnetic ordering. Higher intensity sources are needed for the study of large biological molecules and the investigation of transient samples. The newly developed time-of-flight method holds great promise with both intense pulsed reactors and low flux reactors.

Many dynamical properties of condensed matter have been revealed through inelastic neutron scattering and these measurements of dispersion relations, frequency distributions, anharmonic modification of dispersion relations, Kohn anomalies, etc. will be extended to more complex samples. As the samples become smaller and the structure of the samples become more complex, more intense sources of neutron will be needed to provide better resolution. The neutron methods hold great promise for throwing more light on the microscopic properties of liquids and for complimenting infrared measurements for molecules.

While many macroscopic effects of neutron-radiation damage have been determined, the need is for more precise definition of the cause of damage and microscopic observations of this damage. More intense neutron sources can help to meet both these needs. With more intensity, future experiments could be with monoenergetic beams outside the reactor. Neutrons could also be employed to observe the damage on a microscopic scale.

VI. ACKNOWLEDGEMENTS

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FIGURES

1. Chronological outline of the highpoints of investigations of the physical properties of condensed matter using neutrons.
2. Computer plotted stereoscopic picture of ammonium sulfate obtained in a neutron diffraction experiment.
3. The MTR time-of-flight neutron diffractometer.
4. Example of time-of-flight neutron diffraction data for $\alpha\text{-Fe}_2\text{O}_3$.
5. Parts of the dispersion relations for NaCl.
6. Comparison of frequency distribution measured for vanadium.
7. The mean squared displacement (\times a constant) of sodium atoms in the liquid at short times.
8. The "dispersion relations" for liquid lead.
9. Data showing the direction observation of the excitation of the torsional oscillation in ethane gas.
10. The scattering law for deuterated methane gas.

Electromagnetic Radiation

1912	X-Ray Crystallography
~1930	Infrared Spectra

Neutrons

	Research Equipment	Structure	Dynamical Properties	Radiation Damage
1932	Neutrons Discovered			
1933				
1934				
1935				
1936	ReBe Source And Moderation	Diffraction		
1937		Magnetic Scattering		
1938				
1939				
1940				
1941				
1942				
1943	First Reactor			Wigner Defects
1944				
1945				
1946				
1947		Crystal Diffraction		
1948	Double Axis Spectrometer	Powders, Single Crystals, Light Atoms		
1949		Antiferromagnetic Structure		Thermal Spike Defects
1950	Cold Neutron Chopper	Gases And Liquids		
1951				
1952	In Pile Cryogenic Sample			Defect Annealing
1953				
1954			Placzek And Van Hove Theory	Displacement Spike Defects
1955	Triple Axis Spectrometer		Acoustical Dispersion Relations, Frequency Distributions, Intramolecular Spectroscopy In Solids	
1956				
1957				
1958	Phased Chopper Velocity Selector			
1959			Liquid Helium Dispersion Relations	
1960			Magnetic Dispersion Relation	
1961	Linaos And Pulsed Reactors			
1962			Onset Of Diffusion In Liquids, Kohn Anomaly	
1963				Helium Gas Embrittlement
1964				
1965	TOF Neutron Diffractometer		Dispersion Relations In Liquid Metals, Intramolecular Spectroscopy In Gases	
1966		Diffraction From Proteins	Magnon-Phonon Interactions	

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Figure 1

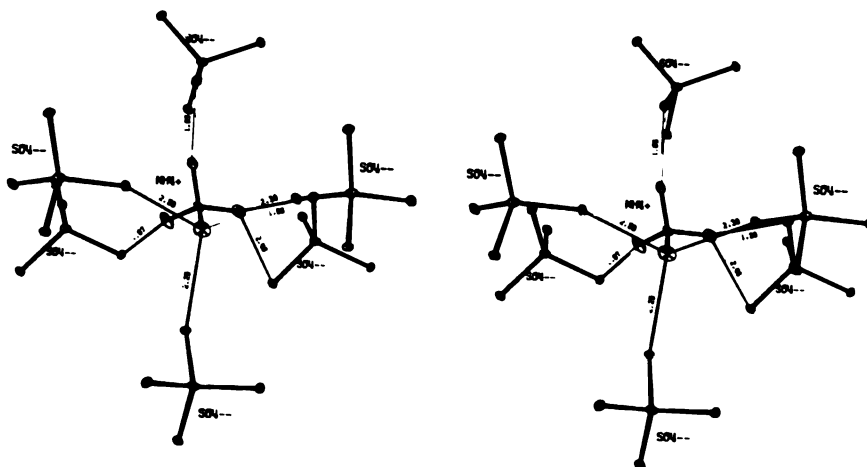


Figure 2

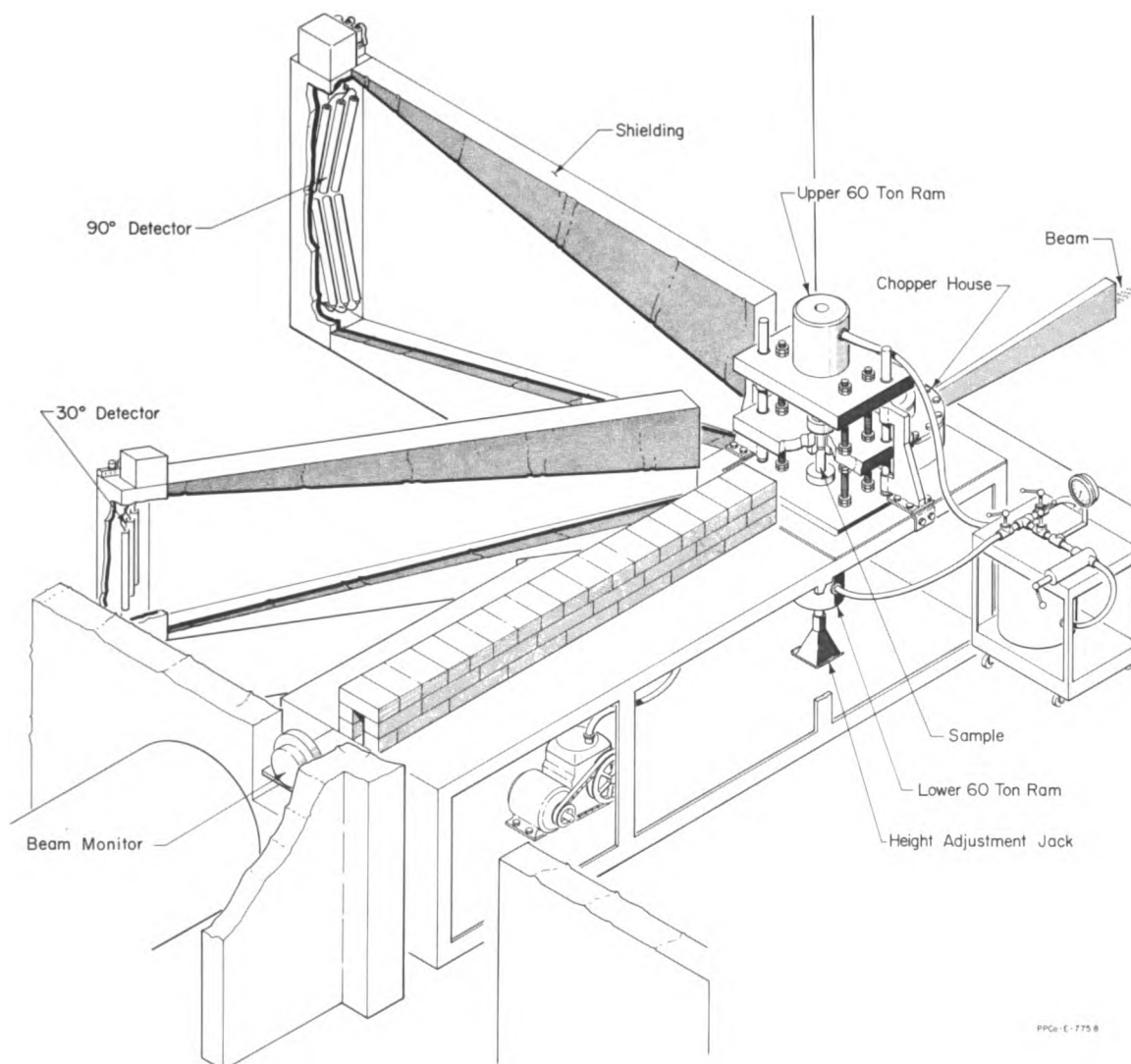


Figure 3

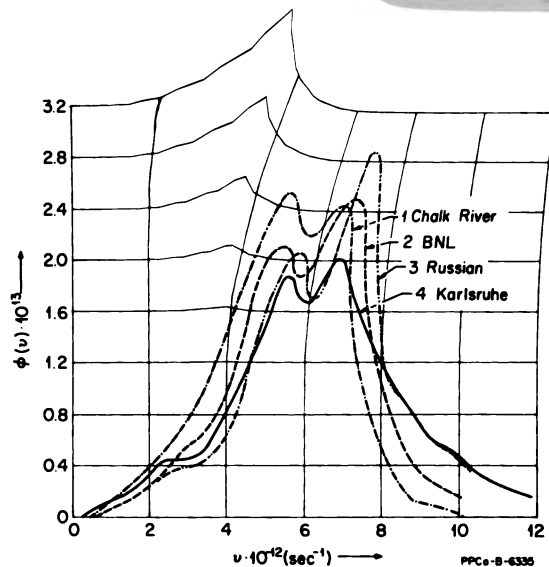


Figure 6

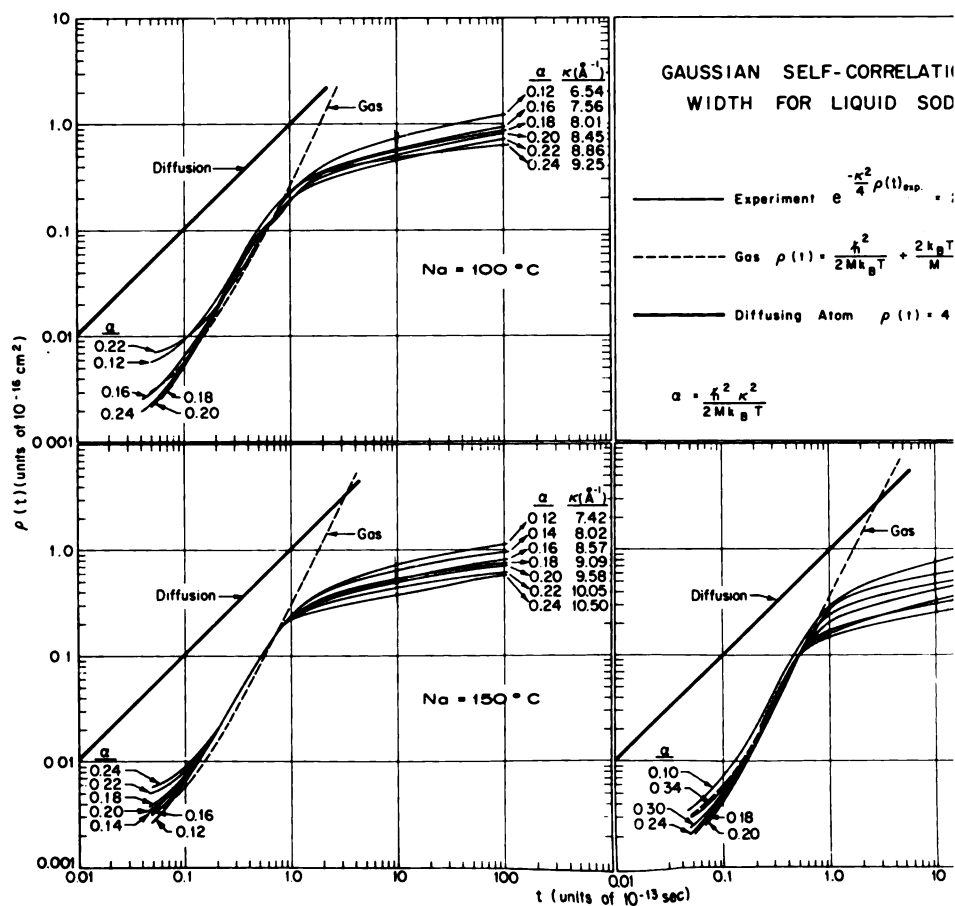


Figure 7

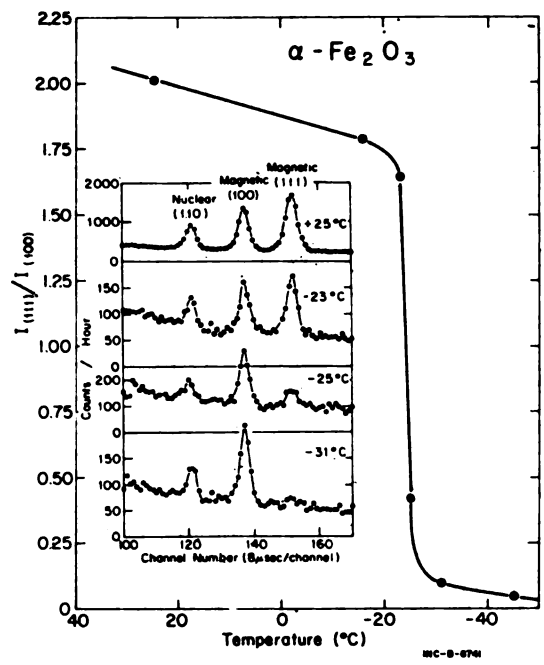


Figure 4

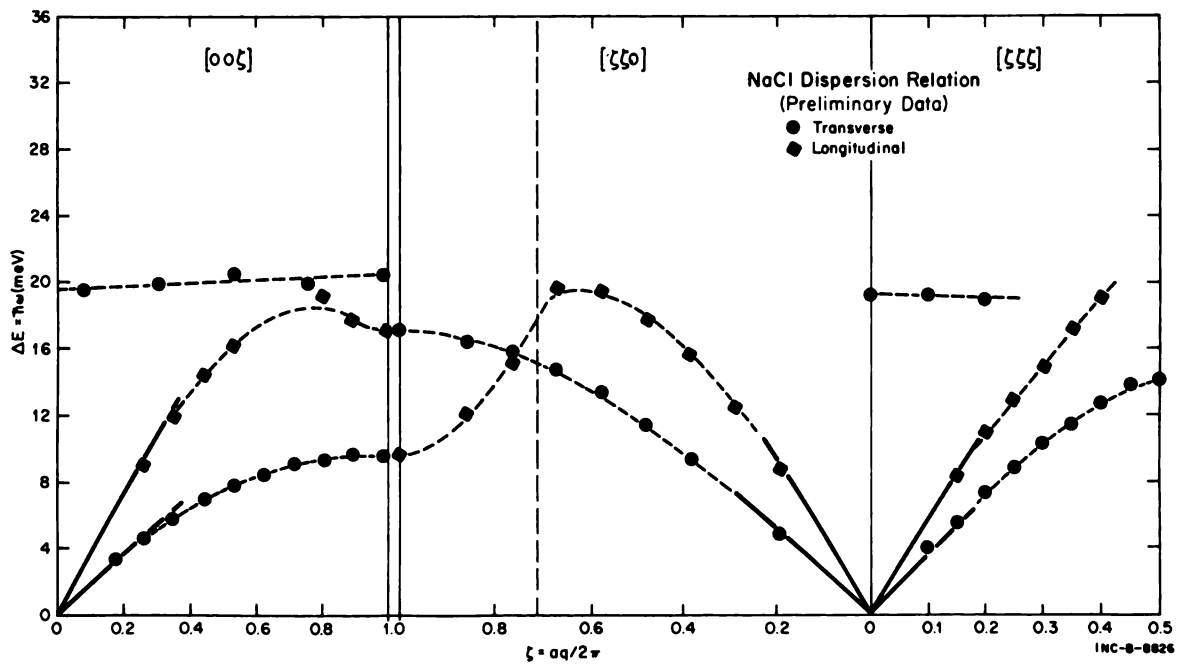


Figure 5

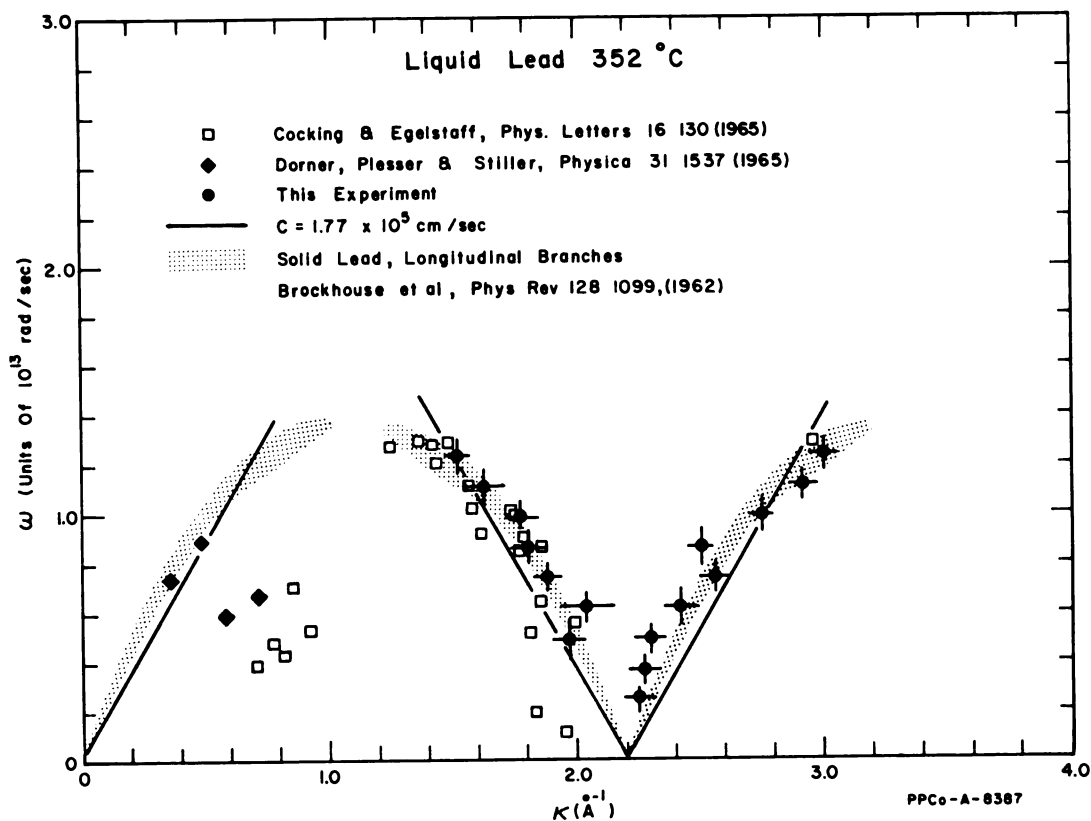


Figure 8

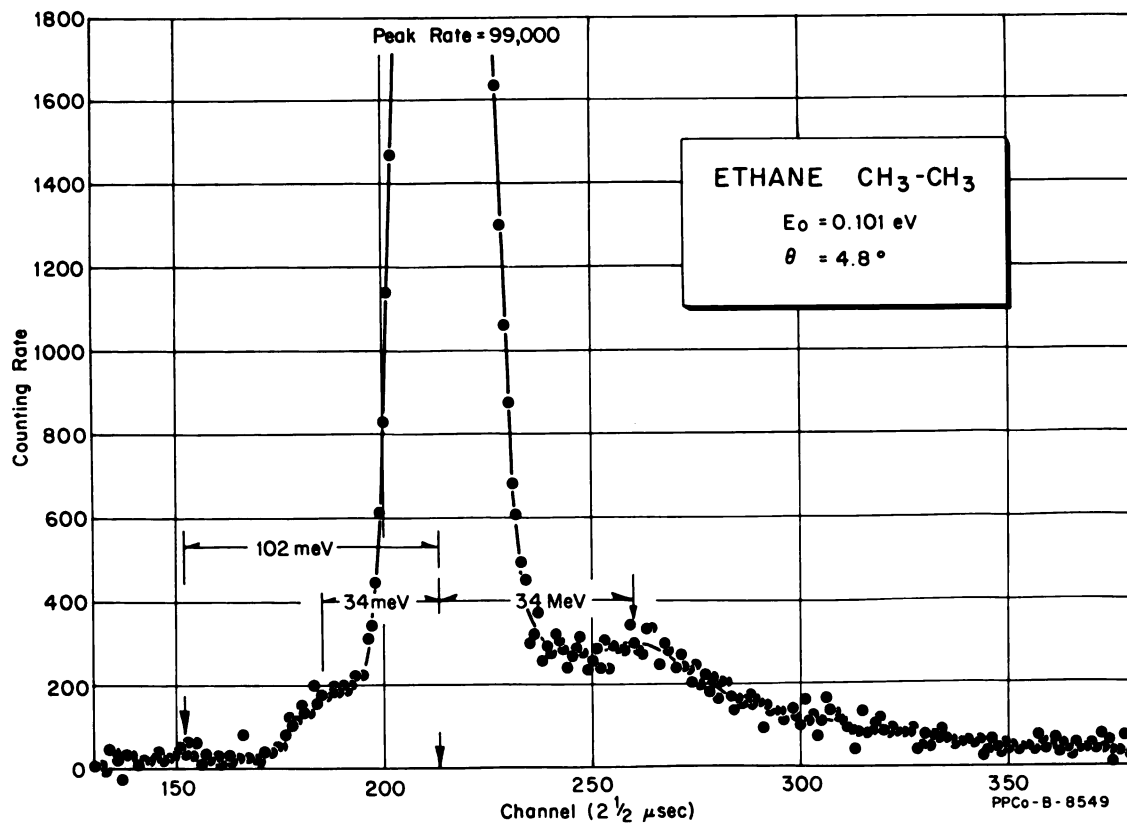


Figure 9

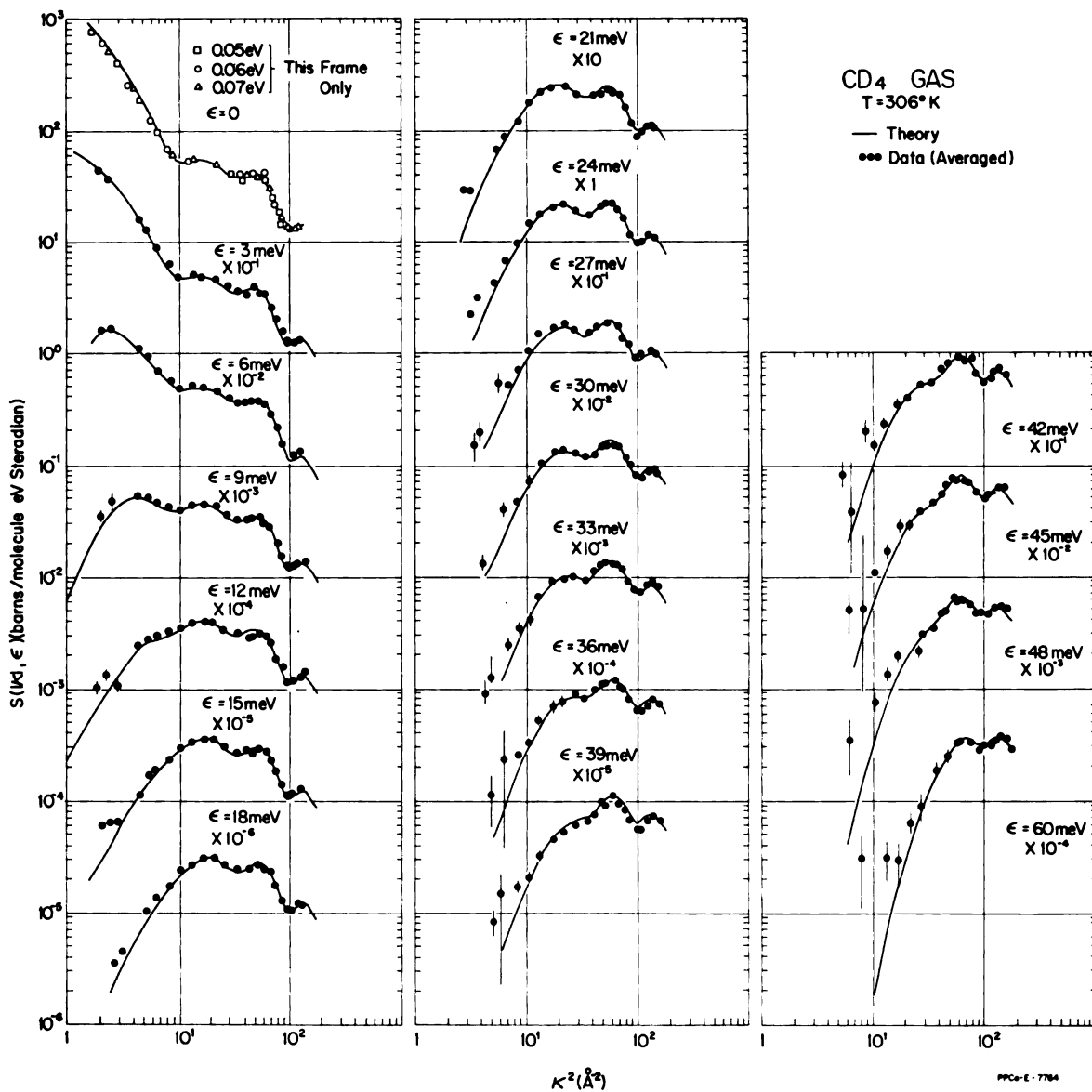


Figure 10

DISCUSSION
OF
PAPER I. A. (BRUGGER)

Chairman: H. Maier-Leibnitz

Secretary: J. Manley

KEEPIN: One point of historical sidelight--in the first figure Brugger showed the time-of-flight method for structure studies as starting in about 1965, and I suppose maybe he was referring to the more complicated crystalline structures, but I believe it is fair to say that the feasibility of the time-of-flight method for structure studies was actually first demonstrated experimentally for simple crystals in 1963 by Buras and also by Buras and Leciejewicz at Świerck in Poland.

BRUGGER: As I mentioned at Dubna, this time is a very difficult thing to determine. If you consider time-of-flight neutron diffraction, you can trace references to it back into the 1950's and even before that. I picked 1965 as a general time when the field started to get exciting, as for many of the other items of the figure.

BEYSTER: I would like to ask Brugger why he felt that the frequency spectra measured for vanadium at various laboratories disagree. Presumably everybody has used about the same method of getting the frequency spectrum from the data, namely, the one-phonon incoherent approximation. Why then are there differences?

BRUGGER: I think one might point out several possible sources of error. One is the treatment of multiple scattering, while another is the treatment of multiphonon scattering. A third source of error is that many people have had very thick samples. No one, except possibly Gläser, has really treated incoherent scattering as a function of momentum change. Experimenters like to take a look at data at just one angle and get some energy changes, then say they know the entire spectrum. My feeling is that, in this inelastic scattering field, you not only treat energy change, but you also treat momentum change since they are coupled.

HARVEY: In your neutron diffraction by the time-of-flight technique, is this advantage mainly for polycrystalline material, or does it also have an advantage for single crystal work?

BRUGGER: At the present time, the work of Buras, that at Idaho Nuclear and several other places, has been mostly with polycrystalline samples. Buras has done some scattering from single crystals. We recently did some scattering on single crystals and it looks like the extinction problems are much more severe for the time-of-flight method than for the variable angle method. However, since they are more severe, they may

have more promise for unusual methods.

MAIER-LEIBNITZ: I think we shall have a discussion later during one of the Panels, or perhaps before, on this special question. It is the best example we have on usefulness of new methods and how it influences the usefulness of various neutron sources.

BURAS: I will present a summary of my paper later, but now I should comment briefly on single crystal work. Lebech and Mikke, at Risø, have studied single crystals of Cr-Rh alloys and the results obtained have shown that the TOF method is also advantageous in the study of single crystals. In regard to the problem of extinction which was just mentioned, this is really a difficult and, for the time being, not a completely solved problem. Recently, we have measured at Świerck the extinction on an aluminum single crystal and we have tried to calculate the extinction by the Monte Carlo method. We have obtained a comparatively good fit of the theoretically calculated with the experimentally measured extinction.

MAIER-LEIBNITZ: Perhaps it is true that for small organic crystals the extinction would not be so much of a problem because they are very imperfect and small. If for the moment I may speak of the use of thermal neutron and high flux reactors, everyone seems to agree that probably neutron diffraction will be a most important application of those instruments in the foreseeable future and that in this application, if we can succeed in seeing the hydrogen location in organic crystals, then this might be a most important field. We should make an effort to overcome the disadvantage in intensity we have compared with x rays, I think it is a disadvantage of 1,000 that we have in luminosity--numbers of neutrons in a 1% energy range/cm²sec and solid angle compared to the same quantity for an x-ray line. The extinction would certainly be an obvious objection to the use of the time-of-flight method, but it would not apply to small organic crystals. Am I not right?

BURAS: I should say yes, but I can not say it for sure because I do not think that anybody has proved it experimentally, but of course, from a logical point of view, the extinction problem should not be so severe for crystals which are not perfect. By the way, I should mention that the problem of extinction in the conventional method is also not solved completely, so this applies not only to the time-of-flight method.

MAIER-LEIBNITZ: Yes, but it becomes more severe because you usually use larger angles and that means longer wave lengths, so the total reflectivity is higher.

CAGLIOTI: With regard to extinction, I think that there is a difficulty with the time-of-flight technique insofar as I believe you get more resolution with this technique where long wave lengths participate. Then, as you know, extinction is a difficult problem the longer the wave length, so this is not too favorable perhaps, unless you have another way to solve the problem.

BURAS: This is perfectly correct; however, as I mentioned just before, the extinction can be most probably calculated using the Monte Carlo method. I am saying "most probably" because we have tried it only on aluminum single crystals, which leaves much work to be done, but judging from this one experiment we think that it is possible. In the course of this Seminar I intend to present some data which will support this statement.

MAIER-LEIBNITZ: Perhaps I could mention a few possibilities which I believe Brugger did not mention. Small-angle scattering and irregularities in lattices were mentioned. It would seem that the neutron method is especially suited to cases of small-angle scattering, because, if you use x rays, there you get problems from double Bragg scattering, or Bragg scattering in one direction and back into all directions. Also, it now seems that you can increase resolution by using neutron back-scattering, which in x rays you can not do at exactly 180° but in neutrons you can do very nearly. You can get a resolution for neutrons which is impossible with x rays; so it would be a nice method for looking at imperfection of lattice structure over relatively large ranges. I might mention that we think after preliminary experiments that the resolution you can get in back-scattering from, say a good silicon crystal, is something like 10^{-7} electron volts. I do not know whether anybody has exploited this yet.

Another thing which I would like to have some discussion on sometime involves those questions of the inelastic scattering law. There are two scattering laws actually--coherent and incoherent scattering laws. If you have intense neutron sources, you might distinguish between these two by using polarized neutrons and looking for only that portion of the scattered beam that has been depolarized because this part represents incoherent scattering only. That should be a nice way to distinguish between the two, especially for liquids where otherwise this would be quite difficult. Another method which might be quite useful in some cases, also for diffraction, would be to try to polarize the atoms in the lattice, so that, in principle, one has an organic crystal and one can polarize the protons at low temperatures. Then you would get a tremendous change in scattering amplitude because you would have a lattice that for the neutrons looks nearly like a lattice of only protons because of their tremendous scattering cross section. Also, I might mention that it now seems possible, either by using very slow neutrons down to 5 m/sec or by using high energy resolution, to see the changes that come about if a neutron flips. Then the nucleus flips, too, and if there is an energy necessary for that you have hyperfine structure effects which now it seems possible to detect with neutrons. Those are some of the fields where there might be an extension over what we have just heard.

KOUTS: I wonder if we could have a little more guidance on how much improvement really is useful in this field and on what kinds of experiments. Are possibilities unlimited? Should we try to increase intensities as much as we possibly can, then

keep on going, or are there cutoff points? This is the kind of thing that the reactor designers could use for guidance.

MAIER-LEIBNITZ: I agree that is one of the purposes for which we are here.

BRUGGER: I did not try to estimate the beam intensities needed in the future, but it seems that in most experiments we are working right on the edge of being able to do the experiment because we do not have intense enough beams. Neutron diffraction, I would say, is close to having enough intensity. For example, the protein measurements seem now to require as much programming and effort of interpretation as they do on getting more data.* (*Note added in proof: Wilkinson says this is not the case: Data can be processed in a few days so that the data accumulation rate is still the longest part of the experiment.) The inelastic scattering experiments certainly do not have enough intensity because they correspond to triple-scattering experiment instead of a double-scattering experiment like the structure measurements. For inelastic scattering, we have to have the additional intensity to make up for that third scattering. Let us say that we need a factor of 100 at least for these experiments before we might get to the point where we are spending more time processing the data than we are in taking the data. In the case of tailored beams for radiation damage, we now have mono-energetic beams of 10^7 n/cm²sec and we probably should have beams of 10^{11} - 10^{12} n/cm²sec in order that the irradiation would not take an abnormally long time. If we had a pulse beam of 10^{12} n/cm²sec, then we might be able to see enough radiation damage in each pulse to measure it.

MAIER-LEIBNITZ: Well, one trouble about this high flux business would seem to be that in order to get a factor of two in resolution you usually need a large intensity increase--something like a factor of 8 or, in a very extreme example, maybe 64. It appears to me that, combined with the availability of higher neutron sources, one will also have to make a great effort to improve experimental techniques to produce better results or obtain more useful data. It is worthwhile to spend money in developing and improving techniques just as one invests in the more intense sources themselves.

BURAS: I think in general there are at least two kinds of experiments which would need more intense neutron sources. One kind needs very high resolution, for this you need more intensity. Other kinds of experiments will be experiments of a new qualitative nature which you can not do if you do not have enough intensity. I am not a crystallographer, but I would like to give one simple example for which you need better resolution. It has happened in the last years that many crystals which are thought to be cubic have been shown to have some very small distortions.

In order to see these very small distortions, you need very good resolution. It has been, for example, demonstrated at the IBR reactor in Dubna that for BiFeO₃ you have a very small rhombohedral distortion. Instead of an angle of 90°, you have an angle of 89° and 35 or 36 min. I think that, as nature has shown that such small distortions

are possible, you can expect that there are distortions of only 1 min instead of 35 min. In order to see them you need to have much better resolution. I do not know if crystallographers would agree to this point, but it was meant only as an example by which it is clear that better resolution is needed to have a better understanding of the structure of solids. The other kind of experiments of new qualitative character have been already mentioned by Brugger, for example, the influence of high pressure or external fields on both structure and crystal dynamics.

Brugger mentioned in his talk experiments which cover three fields: structure, lattice dynamics, and radiation damage. Until now they have been treated separately. I think maybe in the future, perhaps ten, twenty, fifty years from now, people would like simply to see the overall problem. For instance, how during a phase transition both the structure and dynamics change. Another example would be the study of structure and dynamic effects during the course of radiation damage in order to gain further understanding of the kinetics of damage. Until now, when we have used a very large dose, it was extended for several days--sometimes months or years. If we could have a very intense pulsed neutron source--let us say, 10^{22} neutrons in one pulse into the sample--in order to study the kinetics of high dose rate processes, this would be interesting.

MAIER-LEIBNITZ: Your first example, I do not think, was one in favor of high flux reactor, but in favor of a better method. I would say that if you used back-scattering together with time-of-flight measurements of neutrons you can get much, much more resolution.

WILKINSON: In answer to the question about needs for higher flux reactors, I think it might be appropriate to point out that even with the reactors that we have today at Brookhaven and at Oak Ridge, i.e., the HFBR and the HFIR, there may not be sufficient neutron intensities to study certain types of biological molecules. For even the simple biological molecules, it takes something of the order of perhaps 25,000 reflections to resolve one of these structures. With the size crystals that are available today, it is only possible to measure somewhere between 100 and 300 reflections a day, even with completely automatic equipment. Therefore, you can see that, even with the very high flux reactors and continuous neutron sources, these experiments actually take the order of three months or more of data taking and this is merely for some of the simple biological structures. The biological structures that people are interested in actually increase in order of complexity by perhaps factors of 10 and 100, so that I believe there is still a need for more intense neutron sources. Now, I might point out that the measurements to which I am referring are associated primarily with the way data are collected today with one detector. Obviously, with a better method of detection, these times can be reduced. As I understand the method that Buras has suggested for single crystal methods using

time-of-flight techniques, it is possible to measure simultaneously more than one Bragg reflection from single crystals, but I might point out that the main reason that this is possible is because he uses detectors which cover a very wide range in scattering space. Now this same approach, of course, is possible with conventional techniques, so that I am not convinced that there is any real advantage in terms of the rate of taking data for a time-of-flight method over the conventional technique. It merely means that up to this point the conventional techniques have not been developed along these lines. I feel that there is no doubt that the time-of-flight technique can be a very powerful tool, one that will be used in specific classes of experiments supplementing the conventional techniques. However, at the present time I do not believe that it is better than the conventional techniques for complicated structure determinations.

JOHN RUSSELL: I would like to make a comment regarding the use of larger detectors with the time-of-flight method of taking data. At the present time at Harwell there is a program underway to develop an area coordinate detector. There also are other programs in other laboratories looking at a similar sort of data-collection system. The principle involved here is to take a very large detector, put it around the scattering sample, and then record not only the arrival time of each neutron, but the angle of scattering. It then becomes possible to record at all angles simultaneously and retain all the angular information. In this way data-collection efficiency is several orders of magnitude higher than in the conventional technique.

MAIER-LEIBNITZ: I also feel that if you use non-monochromatic neutrons you just get one more parameter. You have at the same time quite a number of reflections. So far we have not found it possible, when you use a single crystal and the monochromatic beam method, to see how you can get as many data per unit time as with the pulsed method. I am not quite sure yet. We will have committees for the high flux reactor that will have to discuss this very soon, but that is my feeling now.

NAGLE: I would like to ask Maier-Leibnitz to elaborate a little bit on his remark that hyperfine structure effects should be observable in neutron scattering. In particular, what advantages would this method have over the more conventional ones with optical, μ -mesic x-ray, recoilless emission, absorption of gamma rays, atomic beam methods, and so on?

MAIER-LEIBNITZ: In principle, one way it can be done is by using back-scattering of neutrons and selecting a neutron energy spread of something like 10^{-7} eV. That is about the lower limit. Now, if you scatter those neutrons incoherently there would be energy changes, if, say, you have a magnetic field or some quadrupole energy, and this energy change can be detected by using the second set of crystals which analyzed the beam. The way of analyzing it is by moving one of the crystals

so you get an energy shift in your primary beam, and so you get a spectrum which is in a way similar to the Mössbauer method. Now your question, "Will this neutron technique be better?" I would say no whenever you can do the Mössbauer effect. It would not be better whenever you can do the nuclear magnetic resonance or paramagnetic resonance. It would work better only in cases when for some reason you do not have a Mössbauer nucleus or your resonances are too broad to easily see them in other resonance work, then you might be able to see them better with neutrons.

NUCLEAR PHYSICS EXPERIMENTS PERFORMED WITH NEUTRONS*

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My assignment is to discuss the bases for an interest in higher fluxes of neutrons from the point of view of nuclear physics. I interpret this to mean literally that I should devote most of my time to a consideration of the need for more intense sources than are now available. As a result, although I will need to give a brief outline of the numerous kinds of experiments for which neutrons are needed, this paper is not a general review of nuclear physics performed with neutrons.

A further limitation on the scope of this paper on "nuclear physics" is imposed by an effort to avoid a serious overlap with Harvey's paper on "neutron cross sections." Most nuclear-physics experiments with neutrons can be said to be measurements of neutron cross sections and whether an experiment is nuclear physics or not depends on the interpretation of the data or on the motivation behind the experiment, not on the measurement itself. Thus, there was no altogether logical way for Harvey and me to divide up the subject matter. As it has turned out, we are both talking about nuclear physics and we are both talking about cross sections; hopefully we will talk about different aspects of nuclear physics and cross sections.

There are several alternative ways in which one could organize the subject matter that needs to be discussed. One approach would be to consider various physical ideas and to explore what

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measurements provide information about these ideas. Another approach would be to consider what experiments could be done with various sources. I reject the first approach on the grounds that it is too far removed from the technological matters that are being emphasized in this conference and I reject the latter because it is too closely tied to the characteristics of existing neutron sources. Instead, I will take the middle ground of considering various classes of measurements, giving a brief indication of the physical significance of the results obtained and a discussion of the extent to which the measurements are limited by the intensities of available neutron sources.

A. Fundamental Properties of the Neutron and Fundamental Symmetries

Under this heading we consider experiments on the free neutron itself or experiments in which the neutron interacts with very simple nuclear systems such as the proton. Also, if they involve fundamental symmetries, experiments on more complex nuclear systems are included. Let us start the discussion by listing some of these experiments and why they are of interest.

1. Lifetime of the neutron for beta decay. An accurate value of this quantity is needed because a complete theory of beta decay must predict the lifetime. A comparison of the lifetimes of the neutron and the muon furnish one of the sharpest tests of any general theory of weak interactions.

2. Asymmetry properties in the beta decay of polarized neutrons. The asymmetries observed (Bu 1960) provide the best available measure of most of the fundamental parameters of the weak interaction (the amplitudes and phases of the scalar, vector, tensor, and axial tensor interactions) without complications arising from nuclear-structure effects. Also, one of these asymmetries furnishes an important test of time-reversal invariance.

3. Electron-neutron interaction. Measurements (Kr 1966) of this quantity give the most reliable information about the slope of the electric-structure factor of the neutron at zero four-momentum transfer. Along with the charges and magnetic moments of the nucleons and a few parameters derived from high-energy electron-scattering

experiments, measurement of the electron-neutron interaction provides the main experimental basis for present ideas about nucleon structure.

4. Tests of CPT symmetries. A measurement of an extremely small asymmetry in the angular correlation of certain gamma rays from capture of polarized neutrons by Ti^{49} (or several other nuclides) would be a sensitive test of time-reversal failure, particularly in the form suggested by Bernstein, Feinberg, and Lee.

5. Test of the conserved-vector-current theory of weak interactions. A measurement of an extremely small asymmetry in capture gamma-ray emission following capture of polarized neutrons by Cd^{113} is required for this test. Conflicting results have been obtained in several experiments of this kind (Ab 1964, Fo 1965).

6. Neutron electric-dipole moment. A direct measure of this quantity can test several theories of time-reversal invariance. The only completed measurement (Sm 1957) to date was made at Oak Ridge in 1950. It set a small upper limit. Because of the intense interest in time reversal now, experiments on the electric-dipole moment have been started at Oak Ridge, Brookhaven, MIT, and perhaps elsewhere. The goal of these measurements is to detect the moment if it is as large as 1% of the previously measured upper limit.

7. Neutron charge. Direct measurement of the neutron charge shows that it is less than 10^{-17} of the electron charge (Sh 1966). However, to test certain theoretical speculations it would be desirable to push the limit down much lower, say to 10^{-20} or 10^{-21} .

8. Gravitational constant. The gravitational constant of the free neutron has been measured (Da 1965) to be the same as that of normal matter to an accuracy of one part in 300. In view of our general ignorance about the gravitational properties of unbound primary particles, a much more accurate measurement would be of interest.

9. The neutron-neutron interaction. A good measurement of this fundamental quantity is needed to test the hypothesis of the charge symmetry of nuclear forces. The neutron-neutron interaction has never been measured directly, although it has been inferred from final-state interactions in nuclear reactions.

10. The neutron-proton interaction. Studies of this interaction have provided us with much information about nuclear forces.

Now let us consider somewhat more fully several of the above experiments that have an obvious need for much higher fluxes of neutrons. One of these is the study of asymmetries in the decay of polarized neutrons. A schematic representation of the experiment performed by Ringo and associates (Bu 1960) at Argonne is given in Fig. 1. A steady beam of thermal neutrons from the reactor CP-5 is reflected from a cobalt mirror to produce the polarized neutrons used in the investigation. These polarized neutrons pass through a large chamber where a small fraction of them decay in the reaction $n \rightarrow p + e + \nu$. The experiment consists of measuring the correlations between the neutron spin, the beta ray, and the recoiling proton.

The principal difficulty in this experiment is that, because of the long lifetime (12 minutes) of the neutron, the fraction of the incident neutrons that decay within the sensitive volume of the counting chamber is extremely small. For example, the fraction of thermal neutrons that decay within a 10 centimeter path length is only about 5×10^{-8} . Thus, if the total flux of polarized neutrons is 10^8 sec^{-1} , the neutron decay rate is only 5 sec^{-1} and, taking into account solid angles and detector efficiencies, the counting rate in the various correlation measurements is only about 10^{-2} to 10^{-1} counts/sec. Anyone who has attempted to do precision measurements with such low rates can appreciate the need for 100 or 1 000 times more neutron flux. Even the new generation of high-flux reactors will not satisfy this need.

Another experiment that would benefit from an almost unlimited increase in neutron intensity is a direct measurement of the neutron-neutron interaction. As mentioned previously, this measurement has not yet been attempted. However, various schemes for its measurement have been suggested. Muehlhause (Mu 1963) has suggested that it is feasible to observe self-scattering by a neutron gas within a straight-through beam hole in a conventional reactor and Fluharty (Fl 1965) has discussed a measurement in which there are

interactions between synchronous pulses of thermal and fission neutrons from a pulsed reactor. A schematic representation of the first of these proposals is given in Fig. 2. Here the collimation is carefully designed to insure that the neutron detector views only the neutron gas within an evacuated tube that passes through a high-flux region of a reactor. Thus, for an ideal system, the detectors will register counts only when there is neutron-neutron scattering within the neutron gas.

Let us estimate the counting rate in the proposed experiment under the assumption that the thermal-neutron flux in the through tube is $10^{15} \text{ sec}^{-1} \text{ cm}^{-2}$, the active volume viewed by the detector is 1 liter, and the n-n cross section is 60 barns. Under these conditions the n-n interaction rate would be $3 \times 10^5 \text{ sec}^{-1}$ and, taking into account the solid angle of the detector, the counting rate would be roughly 1 per sec. The worrisome thing about this rate is not that the absolute value is small, but rather that it would almost surely be difficult to detect the small rate of interest in the presence of a much larger background counting rate. One way to attempt to identify the rate from the n-n interaction is to make use of the fact that the background is directly proportional to neutron flux whereas the n-n interaction rate is proportional to the square of the flux. Obviously, a much more intense source of thermal neutrons would be extremely helpful, since it would increase not only the counting rate but also the signal-to-background ratio.

Our final example of experiments termed "fundamental" is the study of the n-p interaction. Unlike the other experiments listed at the beginning of this section, these experiments have been carried out over a wide range of neutron energy. The objective of the experiments is to determine all of the parameters in the n-p interaction. Since this is such a big subject, I must confine myself to an outline of the extent to which our knowledge of the n-p interaction is limited by neutron-source strengths. In this outline I will lean heavily on a recent review by Perring (Pe 1965).

At low energies ($< 20 \text{ MeV}$) the n-p scattering is dominated by the singlet and triplet s-wave phase shifts. The singlet and triplet scattering lengths and the triplet effective range are all

known to good accuracy from an analysis of the zero-energy data, i. e. , from the total cross section, the capture cross section, and the deuteron binding energy. The singlet effective range, on the other hand, is not well known. It is determined from total cross-section measurements over the energy range less than 5 MeV and there is a serious discrepancy in the results obtained from several sets of data. For none of these low-energy measurements is the present accuracy of the data at a limit that is set mainly by the intensity of neutron sources now available.

At higher energies (> 20 MeV) the experimental situation is much less satisfactory both because it is inherently more difficult to obtain a satisfactory neutron source and because more difficult measurements are required to obtain the scattering amplitudes of interest. Even under the assumption of charge independence, which permits the use of some p-p data in the analysis, to obtain all of the phase shifts in the n-p interaction required five neutron-scattering experiments at each energy, e. g. , two differential cross section, two polarization, and one triple-scattering measurements. Because of the difficulty of obtaining intense monoenergetic sources of neutrons at high energy, most of the measurements have been carried out with neutron beams that have a broad spread in energy or with a proton beam impinging on neutrons bound in deuterium. The much more desirable measurements with monoenergetic neutrons incident on a hydrogen target have been carried out at only a few energies in the range 22 to 90 MeV. These measurements at the Harwell cyclotron make use of the time-of-flight method to select a band of neutrons from the broad distribution emitted when energetic protons interact with beryllium.

Some of the high-current, variable-energy accelerators that are being discussed at this meeting should make it feasible to make much more refined and extensive high-energy measurements on the n-p interaction than are possible now.

To conclude this section on "fundamental properties," let me make a general remark on the nature of the experiments. Since the information being sought is fundamental to physics, the experimenter will never obtain final answers. The answers obtained at any given

stage of refinement may be adequate to test some particular theory of interest. However, as long as the study of nucleons remains an active subject of investigation there will be new theoretical questions that require even more refined measurements of the fundamental properties. Thus, although it would not surprise me if certain classes of experiments now performed with neutrons would be of little interest to nuclear physics say 20 years from now, experiments on the fundamental properties of the neutron may be expected to be important for a very long time.

B. Neutron-Capture Gamma Rays

Neutron-capture gamma rays are of interest from at least four easily identifiable points of view. They are studied (a) for the information they give about neutron-capture mechanisms, (b) as a tool to determine the characteristics of initial states formed by neutron capture, (c) as a means of exciting low-energy states for nuclear-structure studies, and (d) as a source of radiation for resonance scattering and other experiments. My treaty with Harvey requires that he treat the first two of these topics whereas I am to cover the last two.

The development of the high-resolution lithium-drifted germanium-diode gamma-ray spectrometer has created world-wide interest in the use of neutron-capture gamma rays to determine energy levels and decay schemes. The News Letter edited by Vervier lists 33 different laboratories as being involved in such studies. The reason for this sudden interest is, of course, that anyone with any reactor now can easily make measurements that give new and useful information about nuclear structure.

Several kinds of data that are used to construct energy-level diagrams are illustrated in Fig. 3. High-energy gamma rays give the energies of low-energy states directly from the difference between the neutron binding energy and the observed high-energy gamma rays. Coincidence measurements help untangle complex spectra. Coincidence measurements are used also in angular correlation and polarization measurements to determine spins and parities of

self-consistent closed loops (the Ritz combination principle) to locate energy levels that are not detected by means of either the high-energy or coincidence measurements.

One lesson that has been learned by many experimenters is that no one of the experimental techniques listed above permits a very complete determination of the characteristic of nuclear-energy levels. On the other hand, when several of the techniques are brought to bear on the study of a single isotope, neutron-capture gamma-ray spectra become a rich source of information. The power of a multiple attack with several techniques is well illustrated by Fig. 4, which summarizes the results of an analysis by Smither (Sm 1966) of all of the available data on thermal neutron capture in Sm^{149} . These data include internal-conversion spectra measured by Bieber et al. (Bi 1962) at Munich and by Groshev et al. (Gr 1963) in Moscow, and high-energy gamma-ray spectra measured by Groshev et al., in addition to Smither's own coincidence spectra, angular correlations, and low-energy spectra observed with a bent-crystal spectrometer. Closely related data from charged-particle reactions and beta decay were also available and these were combined with the capture-gamma data to extend the level scheme of Sm^{150} deduced from the capture gamma-ray data alone (Fig. 4) to higher energies (not shown in the figure).

A significant feature of the level diagram given in Fig. 4 is that the spins and parities of the low-energy states have been determined from the data, without recourse to particular nuclear models. This is especially important for a nucleus like Sm^{150} , which is in a transition region between spherical and deformed nuclei and hence does not exhibit an easily understood spectrum of levels. This unbiased determination of spins and parities is possible, of course, only because the data from several kinds of measurements were available.

From the point of view of this conference, a key consideration in a discussion of thermal-neutron-capture gamma rays is that neutron sources of adequate intensity are now available for almost all measurements performed with germanium-diode

spectrometers, i. e. , for most of the measurements now in progress. For example, consider the measurements on single gamma-ray spectra that are performed at the straight-through-beam facility in the Argonne reactor CP-5. With this system a sample mounted near the core of the reactor is viewed by a small germanium-diode spectrometer outside of the reactor, as shown in Fig. 5. Although the flux at the target is low ($3 \times 10^{13} \text{ cm}^{-2} \text{ sec}^{-1}$) compared to that in a modern high-flux reactor, a target of only 10 milligrams of aluminum (a substance generally considered to have a low capture cross section) gives a counting rate that is high enough to form a good spectrum in an overnight run. Thus, one sees that the system is so sensitive that small samples of almost all isotopes can be studied effectively. Since coincidence measurements on capture gamma rays must be made with an external beam of neutrons, high counting rates are not so easily available as in the single gamma-ray measurements mentioned above. Nevertheless, for a large fraction of targets the basic limitation in the counting rate is set by the characteristic of the germanium-diode spectrometer, not by the flux available from a high-flux reactor.

One might well ask, then, whether there is any need for higher intensities of thermal neutrons for the study of capture gamma rays. The answer is a definite "yes," since higher fluxes would allow high-resolution magnetic and diffraction spectrometers to become much more effective. Perhaps the most obvious example of the need for higher flux is in the measurement of internal-conversion spectra with a high-resolution magnetic spectrograph. Because of the necessity of using very thin targets in these measurements, they have always tended to require higher fluxes than gamma-ray measurements of the same resolution. This disadvantage will be felt increasingly as the resolution is improved, as it needs to be to resolve the complex spectra. Equally important, much higher fluxes are needed to enable measurements to be extended to most isotopes, since now only a few isotopes with high capture cross sections can be studied effectively. Undoubtedly, these matters have been considered quantitatively by the group who are preparing an internal-conversion spectrometer of advanced design for installation at the Franco-German high-flux reactor.

Similarly, substantial increases in neutron flux would permit the sensitivity and resolution of the bent-crystal diffraction spectrometer for capture gamma rays to be improved greatly. Also, higher fluxes would allow routine use of a system that makes simultaneous use of a diffraction and a germanium-diode spectrometer.

This combination system deserves further comment, since it is an interesting idea that might not be widely appreciated yet. A schematic representation of the system used by Smither (Sm 1965) is given in Fig. 6. The target under investigation is mounted in a straight-through hole in a reactor. Gamma rays from this target are diffracted by a bent quartz crystal. The diffracted gammas are detected by a germanium-diode spectrometer. The principal virtue of the combination system is that it allows one simultaneously to make use of the high resolution of the germanium diode and the high precision of the bent-crystal spectrometer—an advantage gained at a considerable loss in counting rate, of course.

The combination spectrometer is most useful, as compared to other spectrometers, for gamma rays with energies from about 1 to 3 MeV. In this range the energy resolution is better than that of the bent-crystal spectrometer alone. Its advantage in comparison with the Ge diode alone is that, since the diffraction spectrometer limits the energy range of the gamma ray incident on the diode, the pulse-height spectrum in the neighborhood of (for example) the double-escape peaks is not confused by the presence of full-energy peaks or Compton distributions from gamma rays at an entirely different energy.

Some of the same kinds of decay-scheme studies outlined above for thermal neutrons are now beginning to be carried out with germanium-diode measurements on resonance-neutron capture. The most obvious advantage of such measurements is that, by observing the spectra from capture in a single resonance, in effect one makes use of an extremely pure sample. Other major advantages are that capture occurs in a single nuclear state of known spin and parity, the spin and parity can be varied to give additional information, and one can determine the average spectra associated with capture in many resonances of a given kind. A somewhat more subtle advantage is

that the fluctuations from resonance to resonance in partial radiation widths increase the effective resolving power of the gamma-ray spectrometer. These advantages are gained at a sacrifice of many orders of magnitude in the available neutron flux.

An example of an energy-level diagram obtained from resonance capture is shown in Fig. 7. Here on the right we see the energy levels of Pt^{196} obtained by Jackson, Julien, and others (Ja 1966) by means of the time-of-flight neutron spectrometer based on the electron linac at Saclay. The energy levels shown were deduced from the high-energy spectra of about 30 resonances for which the spectra were observed with a 6 cm^3 germanium diode. For comparison, on the left is the energy-level diagram obtained by means of other techniques such as beta decay and the (d,p) reaction. In the case of Pt^{196} , at least, many more levels are observed in resonant capture.

One of the most interesting aspects of the high-energy gamma rays from Pt^{196} is that the behavior of the widths of what are thought to be E1 transitions appear to be anomalous in the sense that the average values of the widths do not depend in any simple way on the gamma-ray energy. In particular, it is noteworthy that no transition to the 1117-keV state is observable, even though the 2^+ assignment of the state is thought to be reliable. If this and other transitions are verified as being E1 transitions, then for the first time one will have high-resolution high-energy gamma-ray data that may be compared with nuclear models without the confusion usually caused by statistical fluctuations in radiation widths. Unexpected effects of the kind that have been observed are most interesting in themselves, of course, but they do tend to make the measurement of average radiation widths a less useful source of information about spins and parities of low-energy states than would be the case if the widths depended on the gamma-ray energy in a smoothly varying manner, as would be expected on the basis of a simple, single-particle model of radiation widths.

At this stage in the development of germanium-diode spectrometers, low counting rate is the major source of difficulty in the measurement of the gamma-ray spectra associated with resonant capture. Therefore, it is important to understand the characteristics

of the neutron sources that may be used in the measurements. In an experiment that requires the examination of many individual resonances, good neutron time-of-flight resolution is essential and the various pulsed sources that excel in this regard are undoubtedly the best sources of neutrons. On the other hand, if the problem of interest requires a refined study of only one or two resonances, as may well be the case for studies of low-energy level structure, then a seemingly powerful pulsed accelerator may be inferior to an old-fashioned chopper or even to a crystal monochromator at a reactor. The reason for this is that, since the minimum flight path used in a time-of-flight measurement is usually fixed by practical considerations, the total intensity of neutrons at a resolved low-energy resonance provided by a chopper or a monochromator may be as great as that from a pulsed accelerator, even though the neutron time-of-flight resolution of the pulsed accelerator is (unnecessarily) very much better. As a result, the pulsed accelerator may well be the ultimate weapon for an attack on resonance-capture spectra, but in the immediate future a great deal of interesting physics will be done with what some people consider to be obsolete tools of research.

Let us turn now to our second major topic on neutron-capture gamma rays, namely, their use as a source of radiation for resonance-scattering and other experiments. The most useful characteristic of such a source is its extremely high intensity. For example, at a reactor with a thermal flux of 10^{15} neutrons $\text{cm}^{-2} \text{sec}^{-1}$ it should be possible to form a gamma-ray beam (outside of the reactor) in which the spectrum is dominated by one or two high-energy gamma rays with an intensity of about 10^{11} high-energy gamma rays $\text{cm}^{-2} \text{sec}^{-1}$. The disadvantage of such a source is that, with the exceptions mentioned below, the gamma-ray energy is not continuously variable.

Several kinds of resonance-scattering experiments have been carried out with thermal-neutron-capture gamma rays (Ar 1963). In one kind, high-energy gamma rays from one material are resonantly scattered by an unrelated target when there is a chance overlap between the narrow incident gamma-ray line and a narrow energy level; in this context "narrow" means a line broadened only by the thermal motion

of the target, i. e. , in the neighborhood of 5 eV wide for heavy targets. In another approach the probability of a chance overlap is increased by using a source in which the high-energy incident line is broadened to perhaps a few keV by the recoil provided by a preceding gamma ray. Variants of these experiments are measurements in which the energy of the incident gamma ray is varied over a narrow range of energy by controlling the velocity of the emitting nucleus. From a physical point of view, a drawback of all of these experiments is that they yield radiation widths for only 1 or 2 energy levels in a range of excitation where only statistical information is of theoretical interest.

A technically similar resonance-scattering experiment (Ha 1965) is one in which a recoil-broadened low-energy gamma ray from capture in a nucleus A is resonantly scattered by a low-energy state in the nucleus $(A + 1)$. The radiation widths obtained in measurements of this kind are of interest to theoretical physics, since widths for individual low-energy states are predicted by various nuclear models. However, background problems have prevented the technique from being generally useful.

Several techniques to provide a gamma-ray source with a continuously variable energy over a wide range of energy have been considered. Axel (Ax 1963) has discussed the scheme in which the energy of a neutron-capture gamma-ray line would be varied by varying the energy of the captured neutrons. In my opinion, neutron intensities are still too low to permit useful measurements with a source of this kind.

Another technique for varying the gamma-ray energy is being developed by Knowles (Kn 1966) at Chalk River. Fig. 8 shows the experimental arrangement. Here the variability in energy is achieved by Compton scattering the radiation from an intense, approximately monoenergetic source such as that provided by thermal-neutron capture in nickel. The gamma-ray intensity is maximized by using a large scatterer that is shaped in such a way as to maintain a constant angle of scattering for relatively large angular apertures. The energy spread of the scattered beam is roughly 2%. The effective resolution of the system can be smaller than this by an order of

magnitude, however, if a germanium diode is used to detect the scattered radiation.

The fundamental drawback of the Compton-scattered gamma-ray source is its low intensity. For example, the intensity of 7-MeV radiation provided by the Chalk River system is about $1 \text{ quantum eV}^{-1} \text{ cm}^{-2} \text{ sec}^{-1}$. Knowles has demonstrated that this is enough intensity for useful measurements, but it is not yet clear to what extent his experimental method is competitive with alternative methods that make use of radiation produced by electrons or positrons from accelerators.

C. Fission

The fission process may be thought of as proceeding through three clearly distinguishable major phases. First, the excited nucleus passes over the saddle point in the potential-energy curve. During the relatively long time that is required to make this transition, the nucleus is "cold," since most of the excitation energy is tied up in deformation, and the system exhibits many of the properties of a stable nucleus. In the second phase, the fissioning nucleus continues to deform until scission finally occurs. During this process most of the simplicity of the saddle-point system is lost as the energy tied up in deformation is fed back into excitation of individual parts of the nucleus. In the final stage, the two (or more) separated fission fragments emit various forms of radiation.

In this brief discussion it is convenient first to consider a class of experiments that give information mainly about the second and third phases of fission. These are measurements of the kinematics of the various radiations from fission induced by thermal neutrons. In them one attempts to determine all of the many interrelationships between the fission fragments, the neutrons, the gamma rays, and the beta rays. For example, a typical experiment consists of studying how the number of neutrons emitted depends on the mass of the fragment.

The experimental techniques used in studies of the kinematics of the fission radiations are numerous and often very sophisticated. However, they do not need to be described here, since the flux of thermal neutrons that is now available is adequate for almost all of the measurements. Rather than a lack of flux, it is almost always the detector or perhaps some part of the electronic apparatus used in the experiment that prevents further improvements in the measurements. The only major class of experiments for which neutron flux is a clear limitation is that in which magnetic analysis is used to separate out the fragments of a given mass (Ew 1963). If enough intensity were available, this technique could be used in a thorough study of the beta decays of fission fragments as a function of the mass and energy of the fragments (Ma 1963).

Many kinds of measurements on the fission radiations would also be interesting at neutron energies higher than the thermal energy, because they might provide us with information about the saddle-point nucleus. In particular, it would be interesting to determine to what extent the characteristics of the various radiations depend on the spin and parity of the initial compound nuclear state, since this spin and parity influence the characteristics of the transition saddle-point states through which fission proceeds. Some experiments of this kind have been performed already. For example, in the initial paper on the concept of the transition nucleus, Bohr (Bo 1955) predicted that the average value of the fission widths depends on the spin of the initial state and the slowly accumulating experimental data seem to be confirming this prediction. Similarly, in their well-known experiments with neutrons from a nuclear explosion, Cowan et al. (Co 1963) have shown that the ratio of symmetric to asymmetric fission varies from resonance to resonance in a way that is consistent with a dependence on the spin of the initial state.

In spite of a few successes such as those just mentioned, most of the measurements of the dependence on neutron energy of the fission kinematics are far beyond our capability because of a need for higher neutron flux. For example, consider an experiment in which we attempt to use an accelerator such as the Nevis synchrocyclotron

to measure the dependence on neutron energy of a relatively simple quantity such as the mass distribution of the fission fragments. Assume that the mass is measured in a very large Frisch-grid ionization chamber. Also, assume that the neutron flight path is adjusted to give a resolution width of 0.1 eV at 100 eV. Then, the total rate of fission in a typical resonance in U^{235} at 100 eV is about 10 counts per day and the rate of symmetric fission is about 10^{-2} per day! Clearly, the neutron flux must be increased by several orders of magnitude before it is worthwhile even to attempt such experiments. On the other hand, it is not clear that such measurements will be of much interest by the time the needed flux is available, since rapid progress in understanding fission is being made by means of experiments that are not limited by neutron flux.

D. Polarization Measurements

Both the physical ideas and the experimental techniques involved in neutron-polarization measurements are very closely related to those for neutron cross-section measurements in general. I choose to emphasize polarization because of the clear-cut need for more intense sources.

Neutron polarization is a subject that is studied over the full range of energy that is generally signified by the term "nuclear physics." At very low energies, where the capture reaction is dominant, the experiment usually consists of measuring the transmission of polarized neutrons through a polarized target. The objective of these experiments is to determine the J values of resonances that contribute to the cross section. The technological problem of producing a polarized target is an inescapable handicap in such measurements, since the transmission of an unpolarized target is independent of the polarization of the incident neutron.

Transmission measurements with polarized neutrons and polarized targets have been carried out mainly by Sailor at Brookhaven and Stolovy at NRL. In both cases the polarized neutrons are produced by reflection from a Co-Fe crystal, a technique that is useful only up to energies of 10 or 20 eV. Recently this limitation has been removed by the development at Dubna of a powerful new

technique in which neutrons are polarized by transmission through a polarized proton target (Sh 1965). This method produces highly polarized neutrons with high efficiency over an extremely wide energy range, as is shown in Fig. 9. Consequently, neutrons from any of the various sources of slow neutrons can now be polarized and the scope of measurements is limited more by the problem of producing polarized targets than by the intensity of polarized neutrons. The extent to which neutron intensity is a limitation is the same as that in any transmission measurement by the time-of-flight method, a subject that lies within Harvey's domain.

At energies that are great enough to allow elastic scattering to be a dominant process it is of interest to determine the polarization of the scattered neutron. As is illustrated in Fig. 10, an experiment of this kind involves a measurement of the left-right asymmetry in the scattering of polarized neutrons from an unpolarized target. Heretofore, the polarized neutrons used in all such measurements have been produced in a charged-particle reaction such as the $\text{Li}^7(p,n)$ reaction shown in the figure. The polarization $P_2(\theta_2)$ of neutrons scattered at an angle $\pm \theta_2$ is determined from the relationship

$$P(\theta_1)P(\theta_2) = \frac{1 - r(\theta_2)}{1 + r(\theta_2)},$$

where $P_1(\theta_1)$ is the polarization of the incident neutron beam and $r(\theta_2)$ is the ratio of the neutron intensities scattered to the left and right.

The results (Ba 1963) of polarization measurements are of interest from several rather different points of view. For the lightest targets such as the proton, polarization measurements are used to obtain information about nuclear forces. For slightly heavier nuclides, which exhibit widely spaced resonances, polarization measurements are useful mainly as a source of spectroscopic information. Since the polarization is very sensitive to interference terms in the cross section, polarization data often reveal features of nuclear structure that are undetectable from the cross sections alone. For heavy nuclides with small level spacings, the measured polarization

is an average effect for many resonances and the data are interpreted in terms of various models of nuclear reactions. For example, the polarization data (in combination with the angular distributions of scattered neutrons) are our principal source of information about the neutron potential well in the optical model. In particular, since polarization is caused by a spin-orbit interaction, the polarization data are an important source of information about the spin-orbit term in the potential.

A current example of how important polarization can be in the understanding of reaction mechanisms is provided by a study of "intermediate structure" observed in the neutron cross sections of fluorine and neighboring elements. The total cross section of fluorine is shown in Fig. 11. The lower curve is the original data obtained with a resolution width of about 20 keV and the upper curves were obtained by mathematical averaging over successively wider intervals. This set of curves shows that the total cross section has a structure with a characteristic width of about 150 keV, a width that is intermediate between that of the fine-structure resonances and the giant single-particle resonances. It has been suggested that this intermediate structure results from an excitation of the "doorway states" that have been postulated by Feshbach and associates (Fe 1965). However, in view of the extremely broad fluctuations of neutron widths (Porter-Thomas fluctuations), there is good reason to worry that the observed intermediate structure in the total cross section is nothing more than a chance fluctuation. To resolve this question, Monahan and Elwyn (Mo 1966) have performed a thorough-going phase-shift analysis of all of the available data, which includes the total cross section, angular distribution, and polarization as a function of neutron energy. All features of the data are found to be consistent with the behavior expected of doorway-state excitations. In particular, each of the broad maxima seen in the figure behaves as though it is formed by a broad state with a definite spin and parity. This interpretation is convincing largely because it is consistent with both the polarization and the cross-section data.

Despite the obvious value of polarization data for the understanding of nuclear structure, relatively few targets have been studied over a wide range of energy. To emphasize this deficiency, I have plotted in Fig. 12 the number of elements studied versus the neutron energy range of the study. One sees that most of the measurements have been restricted to the range 0.2 to 4 MeV and that above 4 MeV only a few measurements have been made at a few isolated energies. Almost all the measurements were performed with large targets of easily obtained elements.

A consideration of the problem of obtaining satisfactory sources of polarized neutrons partially explains why the energy range above 4 MeV has been neglected. In principle polarized neutrons may be produced either in a charged-particle reaction or by elastic scattering of neutrons. However, in practice the only useful sources are charged-particle reactions, since a neutron-scattering process reduces the initial neutron intensity by several orders of magnitude. It is for this reason that almost all polarization measurements have been made with reaction-neutron sources at electrostatic accelerators rather than with the various pulsed sources that originate in the fission or evaporation processes. The Dubna source mentioned previously may be expected to change this situation at energies less than 200 keV.

As far as we now know, few charged-particle reactions are good general-purpose sources of polarized neutrons. The characteristics (Ha 1963) of several of the best are summarized in Fig. 9. Below 2.5 MeV the $\text{Li}^7(p,n)$ reaction yields neutrons with a fairly high degree of polarization and it is widely used. At energies greater than 14 MeV the $\text{T}(d,n)$ reaction also gives high polarization, but it is not used much, presumably because of severe technical difficulties in experiments with high-energy neutrons. And for the intermediate range 2.5 to 14 MeV there appears to be no generally good source, although some reactions such as $\text{Be}^9(\alpha,n)\text{C}^{12}$ produce well-polarized neutrons at particular energies.

In all ranges of neutron energy much higher fluxes of polarized neutrons are required to enable one to correct two obvious deficiencies in present polarization measurements. First, good measurements on individual isotopes should be made. And second, the measurements should be extended to higher energy so as to minimize the correction that must be made for compound-elastic scattering. However, at higher energies the experimental problem of discriminating against inelastically scattered neutrons becomes increasingly severe and a pulsed-beam technique, with its resultant loss in intensity, may be required.

Let us conclude this section on polarization measurements by inquiring whether there is sufficient physical interest in the problems that require polarization measurements to justify a continuing effort to increase the available neutron intensity. Just one example will show that there is. In the study of the optical model, as a minimum objective one would like to have our future knowledge of the neutron potential well be as complete as our present knowledge of the proton well. To achieve this will require a large increase in both the quality and quantity of neutron-polarization data.

E. Triple Scattering

Under this heading we consider experiments intended to measure the Wolfenstein parameters, which, in addition to the polarization and differential cross section, are required to specify the scattering matrix (Wo 1954). Perhaps because almost no experiments of this kind have been performed, there has been no published discussion of the extent to which experimental values of the Wolfenstein parameters are likely to be useful in the elucidation of nuclear structure. Nevertheless, since the parameters are independent of the scattering cross section and the polarization and since even rather complete data on these two quantities would be expected to leave many questions unanswered, it seems obvious that an effort should be made to measure the Wolfenstein parameters. One specific objective of the measurements would be to obtain information about a possible spin-spin interaction for the neutron.

A schematic representation of the kind of measurement needed to determine the Wolfenstein parameters is given in Fig. 13. Note that the conventional term "triple scattering" is somewhat confusing in such a measurement since, although there are three nuclear interactions, there are only two scatterings. In the example given in the figure, polarized neutrons from a $\text{Li}(p,n)$ reaction are incident on an unpolarized target, where they are scattered at an angle θ_2 . The scattered neutrons then impinge on a second unpolarized target and various asymmetries in the intensity of neutrons scattered from this target are measured. In an experiment of this kind, the final scatterer may be thought of as an analyzer to determine the extent to which the polarized incident beam is depolarized by the first scatterer.

My only objective in this discussion of triple scattering is to emphasize the degree to which higher intensities of neutrons are essential if triple-scattering measurements are to become generally useful. Because of the need for an additional scattering, the counting rate for a triple-scattering experiment would be roughly 10^4 times smaller than that for a polarization measurement of comparable accuracy. As a result, polarized neutron beams that are an order-of-magnitude more intense than those used heretofore in polarization measurements will be required before any useful triple-scattering measurements can be made and an increase by several orders of magnitude is needed to allow one to accumulate a set of data such as is now available on polarization.

F. Small-Angle Scattering

This topic is separated from the discussion of polarization measurements, to which it is closely related, because the motivation behind the recent experiments on small-angle scattering has been quite different and because the experimental difficulties in the measurements provide a graphic demonstration of the need for more intense sources of neutrons.

Small-angle scattering is of physical interest because it can reveal non-nuclear interactions that are not detectable by scattering at larger angles. One of these is an interaction between the magnetic

dipole moment of the neutron and the Coulomb field of the scattering nucleus, an effect known as Schwinger scattering. This interaction is expected to cause a large increase in the scattering cross section and a large polarization at angles less than about 5 degrees. In spite of severe technical difficulties, these phenomena have been observed and their expected magnitudes qualitatively verified at several laboratories.

Another effect that needs to be considered in small-angle scattering is a possible electric-dipole polarization of the neutron. Considerable efforts to measure the polarizability has been made by Alexandrov et al. in the Soviet Union and recently by Fossan and Walt and by Elwyn et al. in the United States. An example of the results obtained in such experiments is given in Fig. 14. Here we compare the polarization and differential cross sections measured by Elwyn et al. (El 1966b) with their calculations of these quantities under several sets of assumptions about the nature of the interactions responsible for the small-angle scattering. There are several noteworthy features to the data. First, the maximum polarization is very large, a point to which we shall return later. Second, the data give clear-cut evidence for Schwinger scattering. Third, and perhaps most interesting, the scattering cross-section data are "anomalous" in the sense that they appear to be inconsistent with the calculated cross sections for a value of α , the neutron polarizability, that is smaller than 10^{-40} cm^3 , whereas the polarizability is expected to be less than $2 \times 10^{-42} \text{ cm}^3$.

The evidence concerning "anomalous scattering" at small angles has been summarized recently by Elwyn et al. (El 1966b). The effect has been observed in several laboratories but only for fissionable targets and only at energies greater than the fission barrier. Also, the effect is not observed at all energies greater than the fission barrier. Since there is no satisfactory theoretical explanation of all of the existing data, one is led to consider the possibility that some long-range interaction of unknown origin influences the small-angle scattering. On the other hand, in view of the technical difficulties in the measurements, perhaps one is justified in being skeptical of whether the observed discrepancies between theory and experiment are significant. Nevertheless, since the present discrepancies have

raised questions of fundamental importance, much more extensive and accurate measurements are highly desirable. It is unlikely that they will be made until there is an order of magnitude increase in the neutron flux available for the measurements.

As we have seen, until now small-angle scattering measurements have been aimed at understanding the fundamental processes involved. However, it is possible that small-angle scattering will be of interest in the future mainly as a technique of obtaining highly polarized beams of neutrons for a wide range of energies. One may obtain a qualitative idea of the characteristics of such a source from Fig. 14. For example, if 0.83-MeV neutrons were scattered from uranium within the angular aperture 0.9—1.7 degrees, the polarization would be very large (about -0.75)—but the neutron intensity would be reduced to about 10^{-4} of its initial value. It is likely that it will be a long time before we are rich enough in neutrons to be able to afford the luxury of such a source for routine use.

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Figure Captions

- Fig. 1. Schematic representation of an experimental system to measure correlations in the neutron-decay products.
- Fig. 2. Schematic representation of an experimental system to measure neutron-neutron scattering.
- Fig. 3. Frequently used types of neutron-capture gamma-ray measurements.
- Fig. 4. Schematic representation of a high-sensitivity capture-gamma facility.
- Fig. 5. Energy levels of Sm^{150} deduced from thermal-neutron-capture gamma-ray measurements.
- Fig. 6. Schematic representation of an experimental system that combines a diffraction and a Ge-diode spectrometer.
- Fig. 7. Energy levels of Pt^{196} . For the levels observed in resonance capture, the spin of the final state is $J = 1$ if the initial state has $J = 0$; otherwise, the spin of the final state is 0, 1, or 2.
- Fig. 8. A schematic representation of the Compton-scattering variable-energy photon source at the reactor NRU.
- Fig. 9. Characteristics of sources of polarized neutrons. The data for the charged-particle-induced reactions were taken from a review by Haeberli (Ha 1963).
- Fig. 10. Schematic representation of a typical neutron-polarization measurement. All trajectories lie in one plane.
- Fig. 11. The total neutron cross section of fluorine as a function of neutron energy for various resolution widths ΔE .

Fig. 12. Number of elements for which neutron-polarization measurements had been reported by late 1964. The ordinate represents the number of targets studied at any energy within the indicated ranges of energy (El 1966a).

Fig. 13. Schematic representation of a triple scattering measurement. All trajectories lie in one plane. To obtain different information the final scattering plane should be rotated through 90° .

Fig. 14. The angular distribution and polarization of 0.83-MeV neutrons scattered at small angles by uranium. The parameter α is the neutron polarizability.

ASYMMETRIES IN NEUTRON DECAY

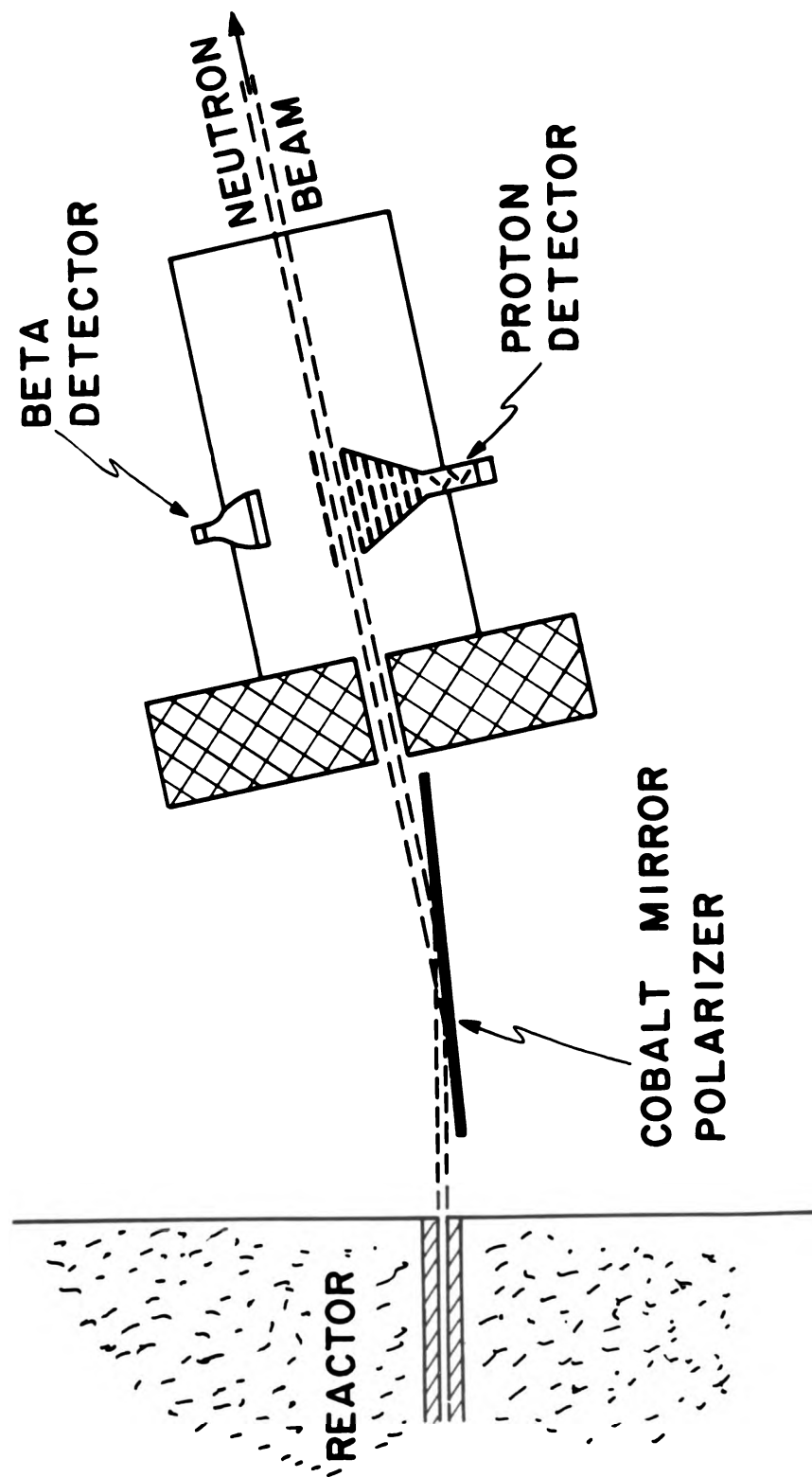


Fig. 1

NEUTRON - NEUTRON SCATTERING

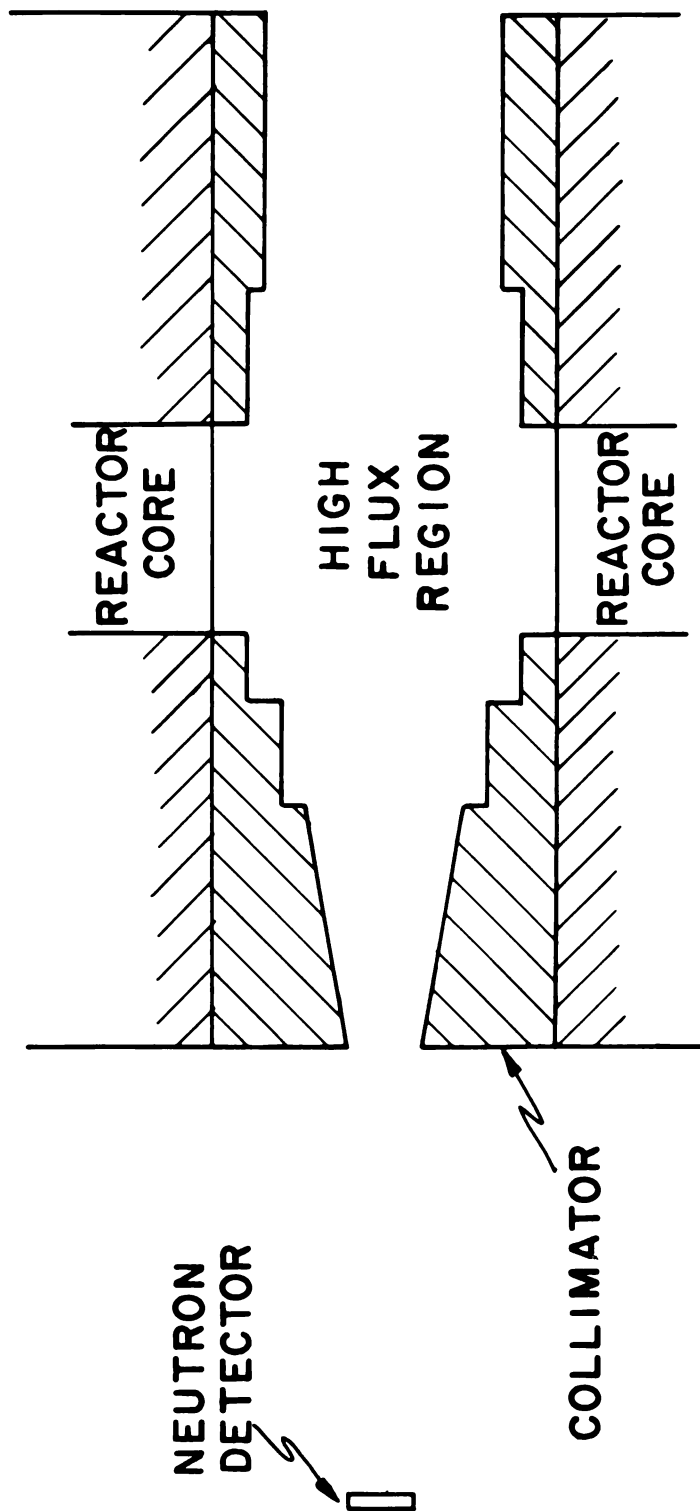


Fig. 2

TYPES OF CAPTURE γ -RAY SPECTRA

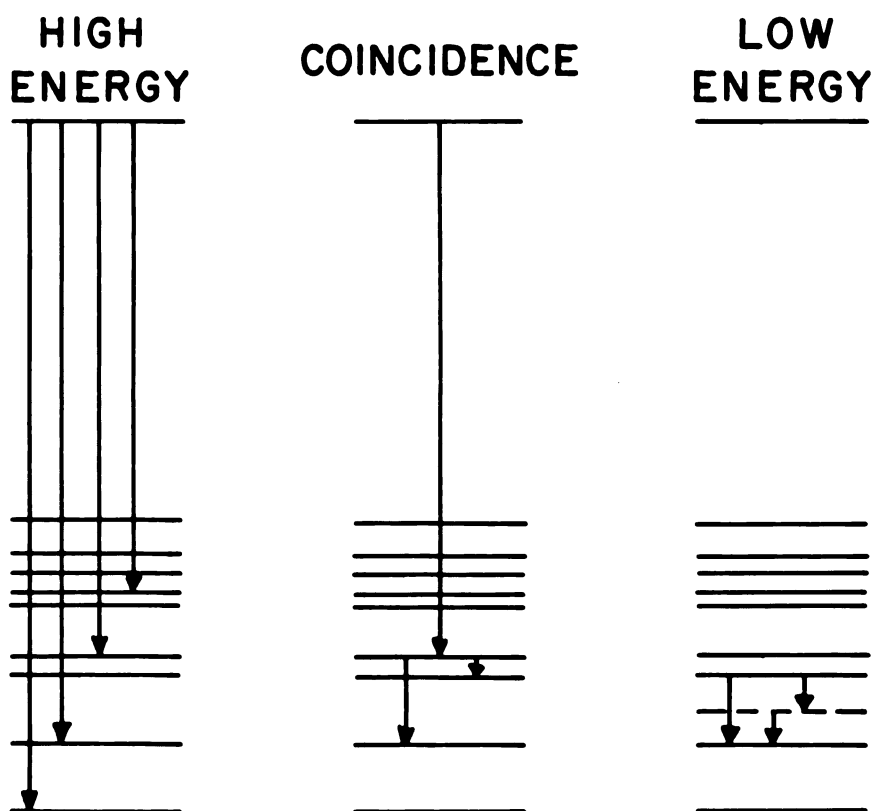


Fig. 3

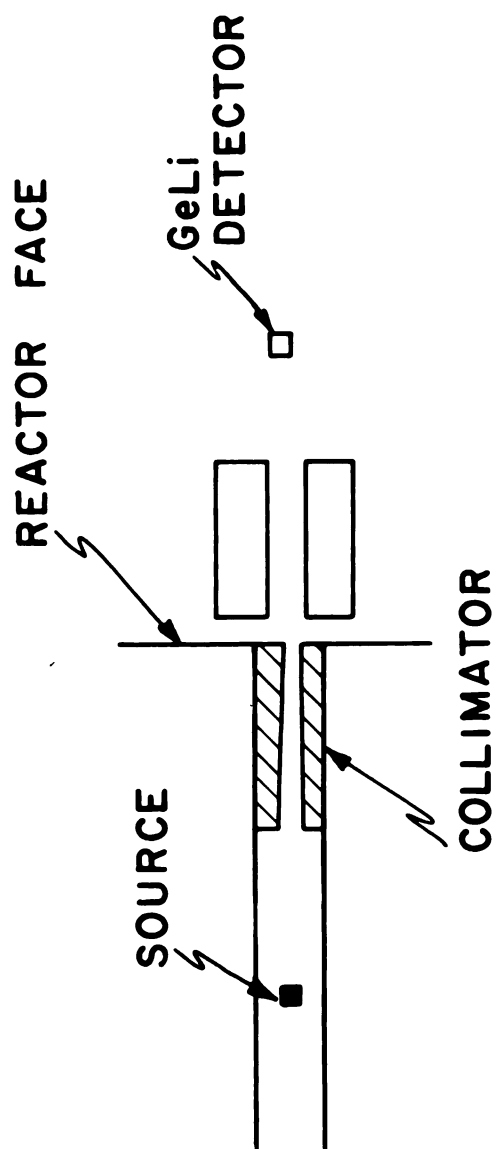


Fig. 4

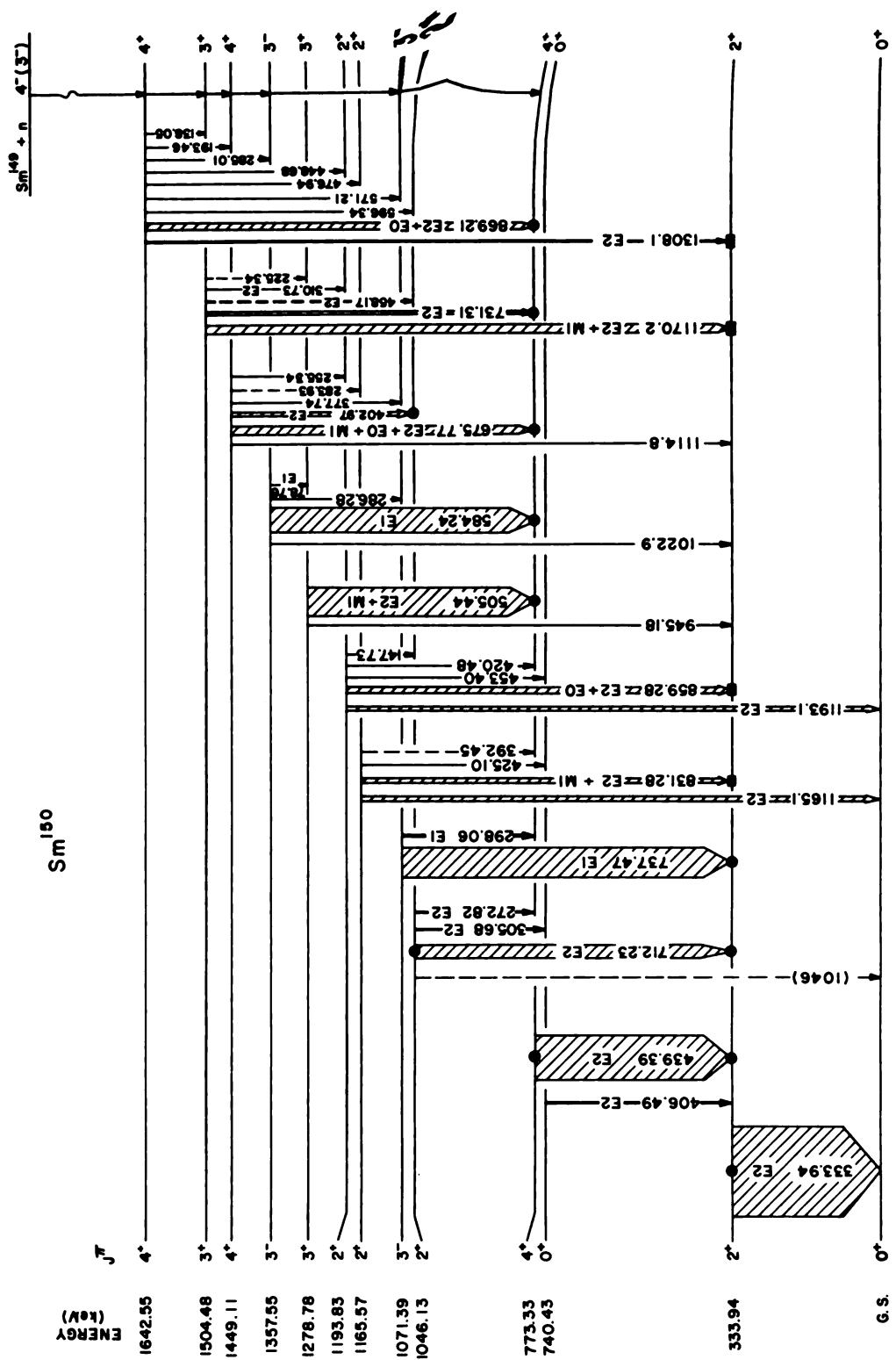


Fig. 5

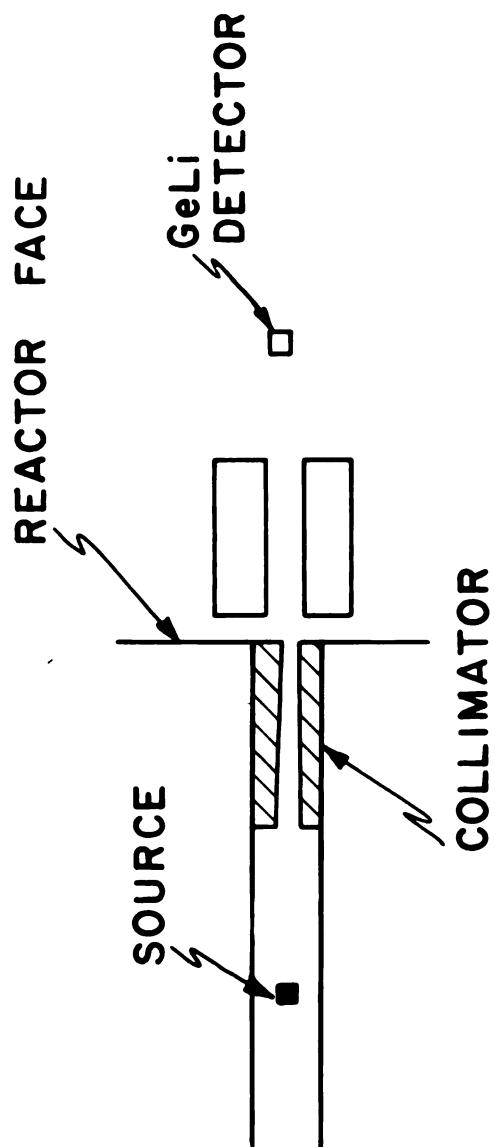


Fig. 4



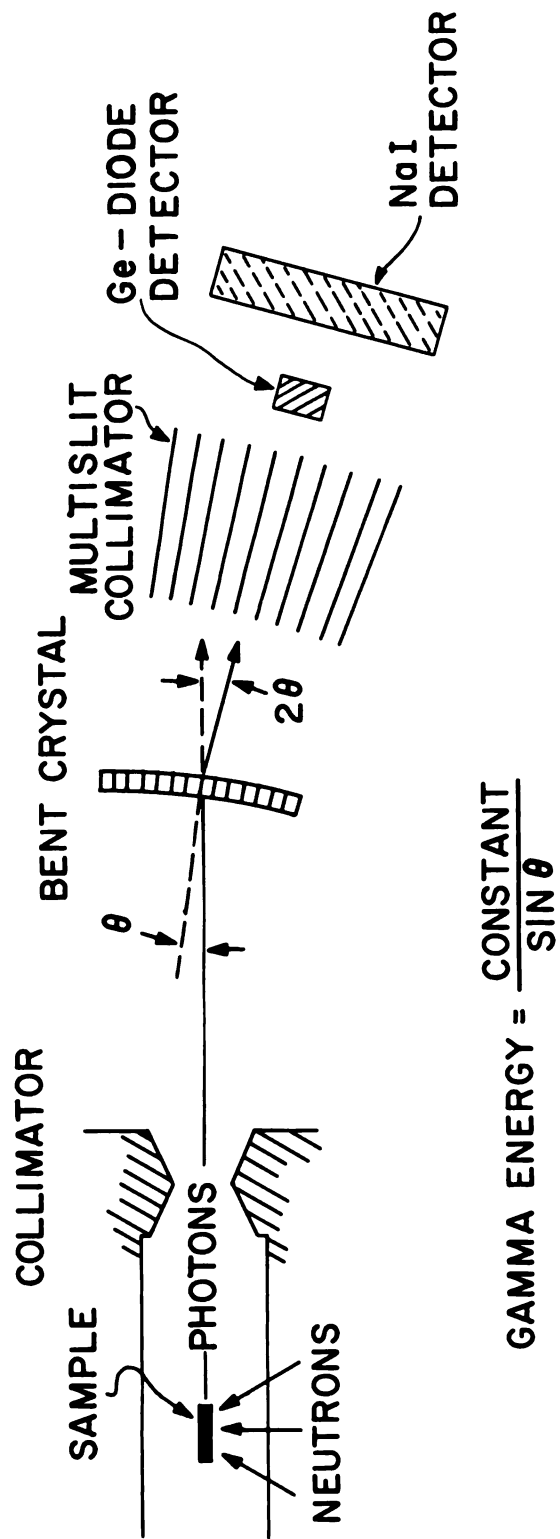
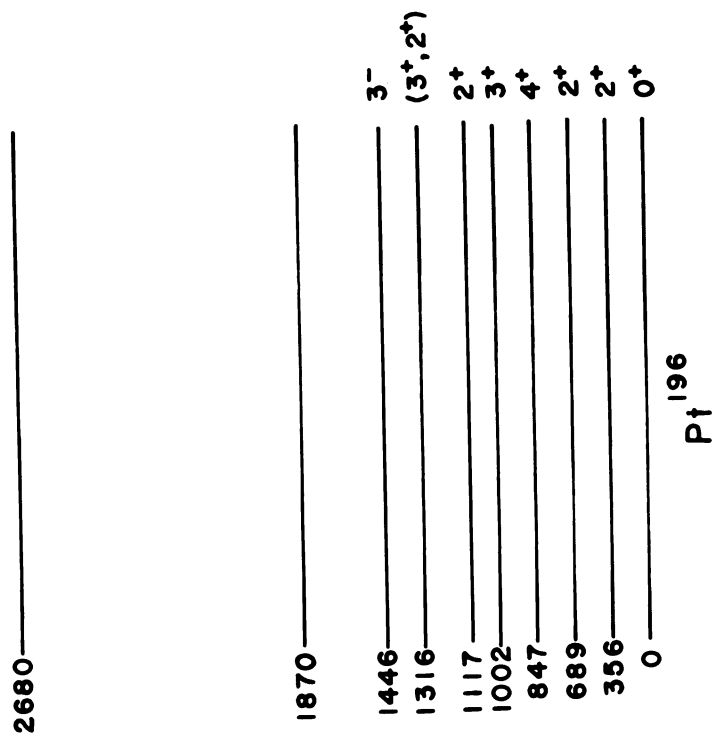
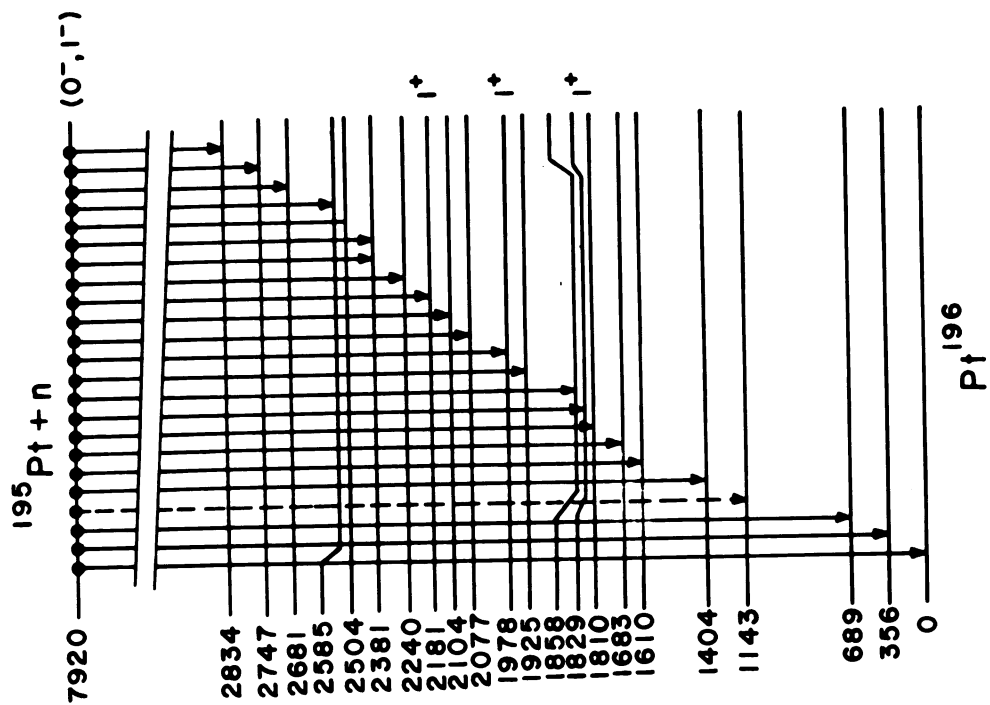


Fig. 6

β DECAY and (d, p)



NEUTRON CAPTURE



COMPTON - SCATTERING VARIABLE - ENERGY SOURCE

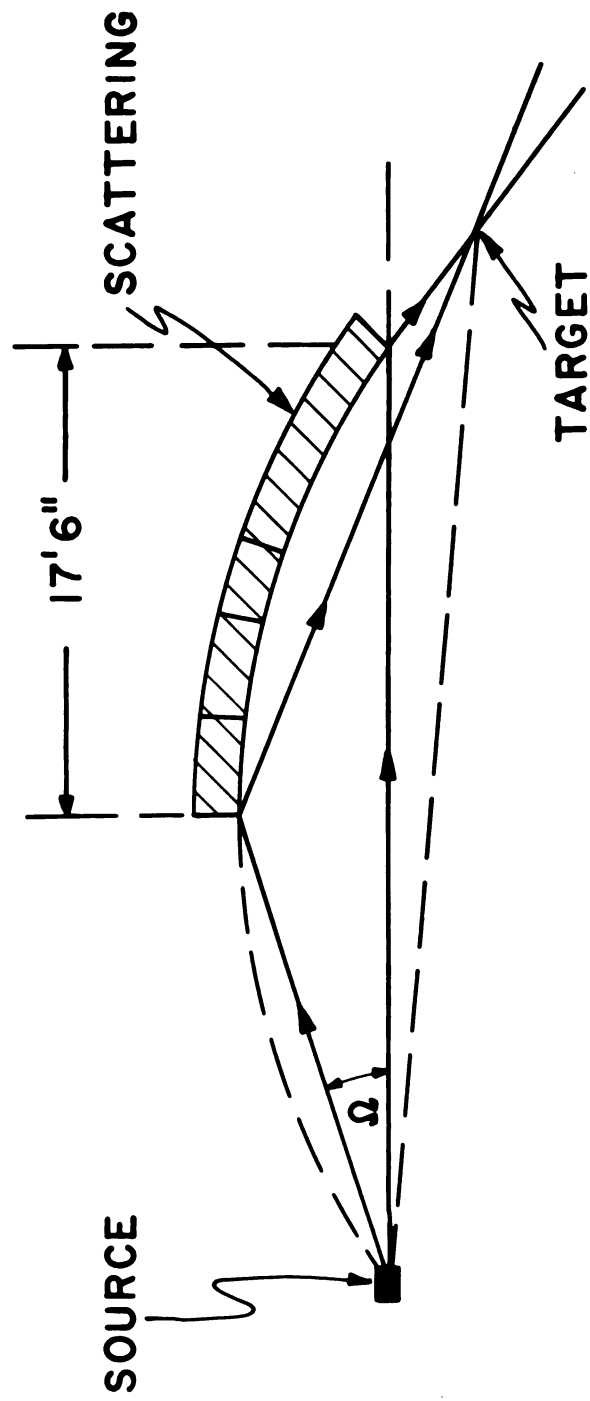


Fig. 8

POLARIZATION MEASUREMENT

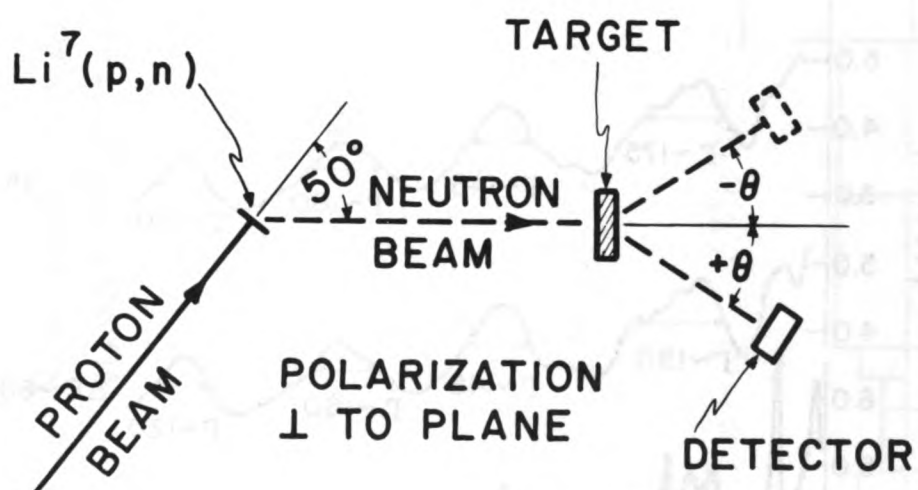


Fig. 9

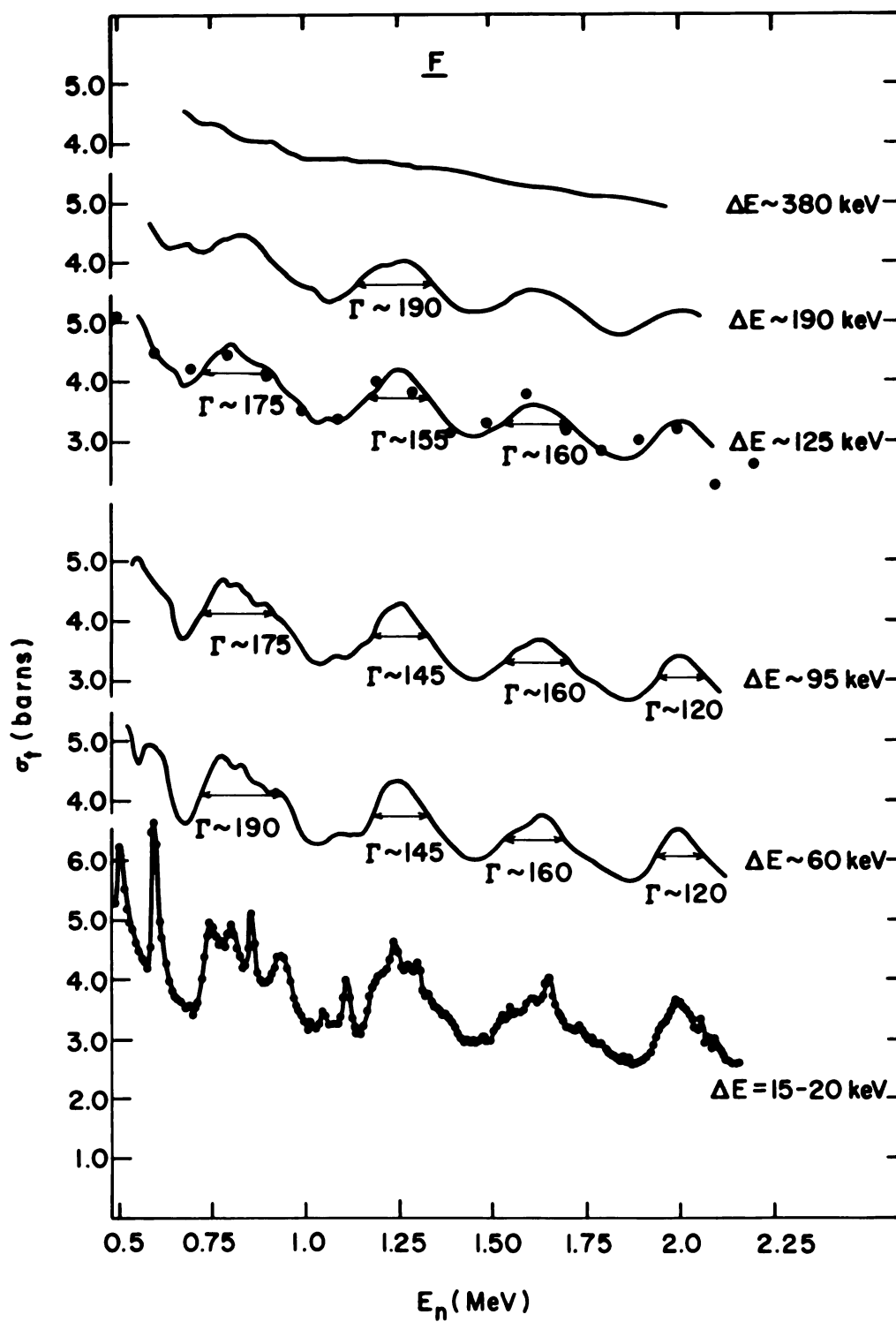


Fig. 10

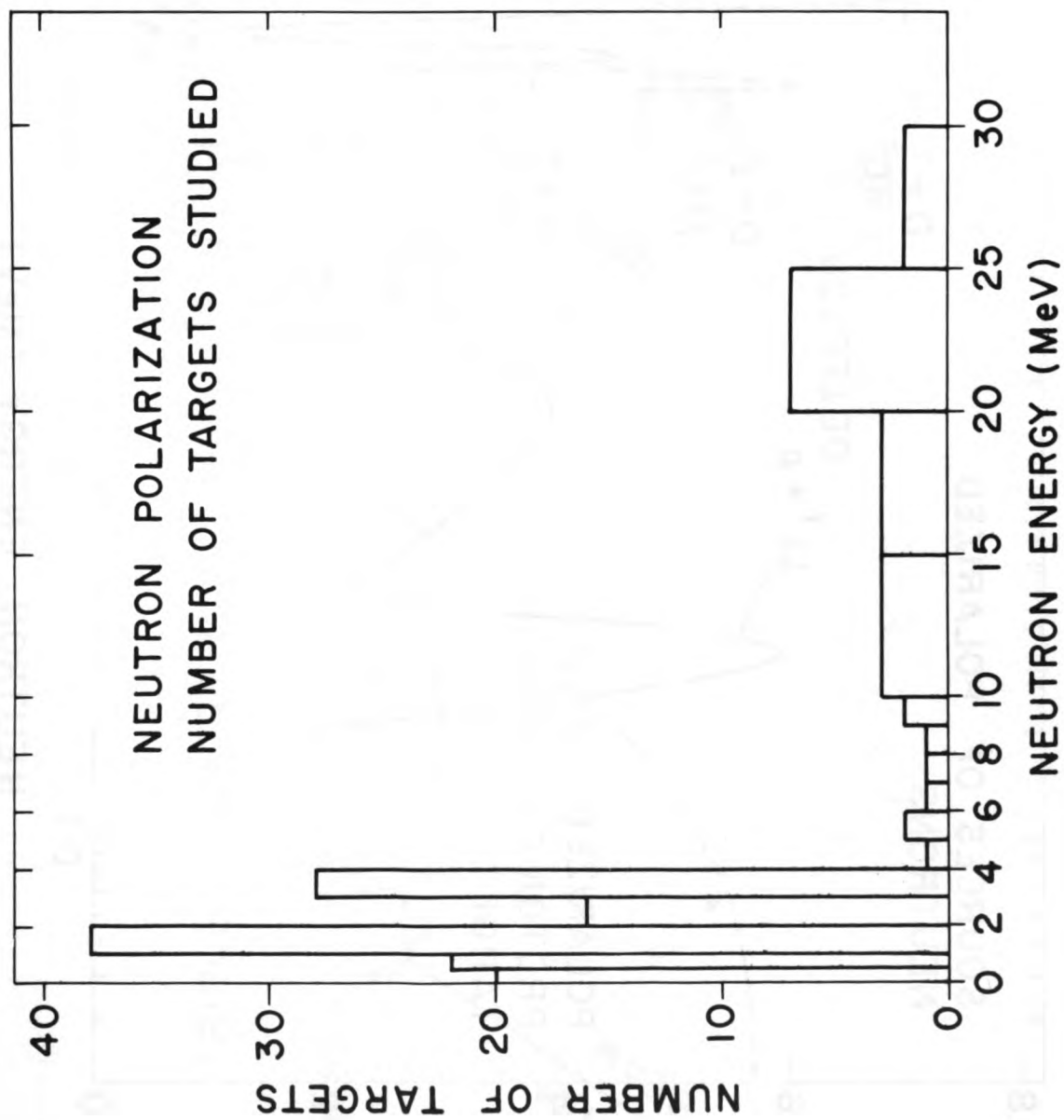


Fig. 11

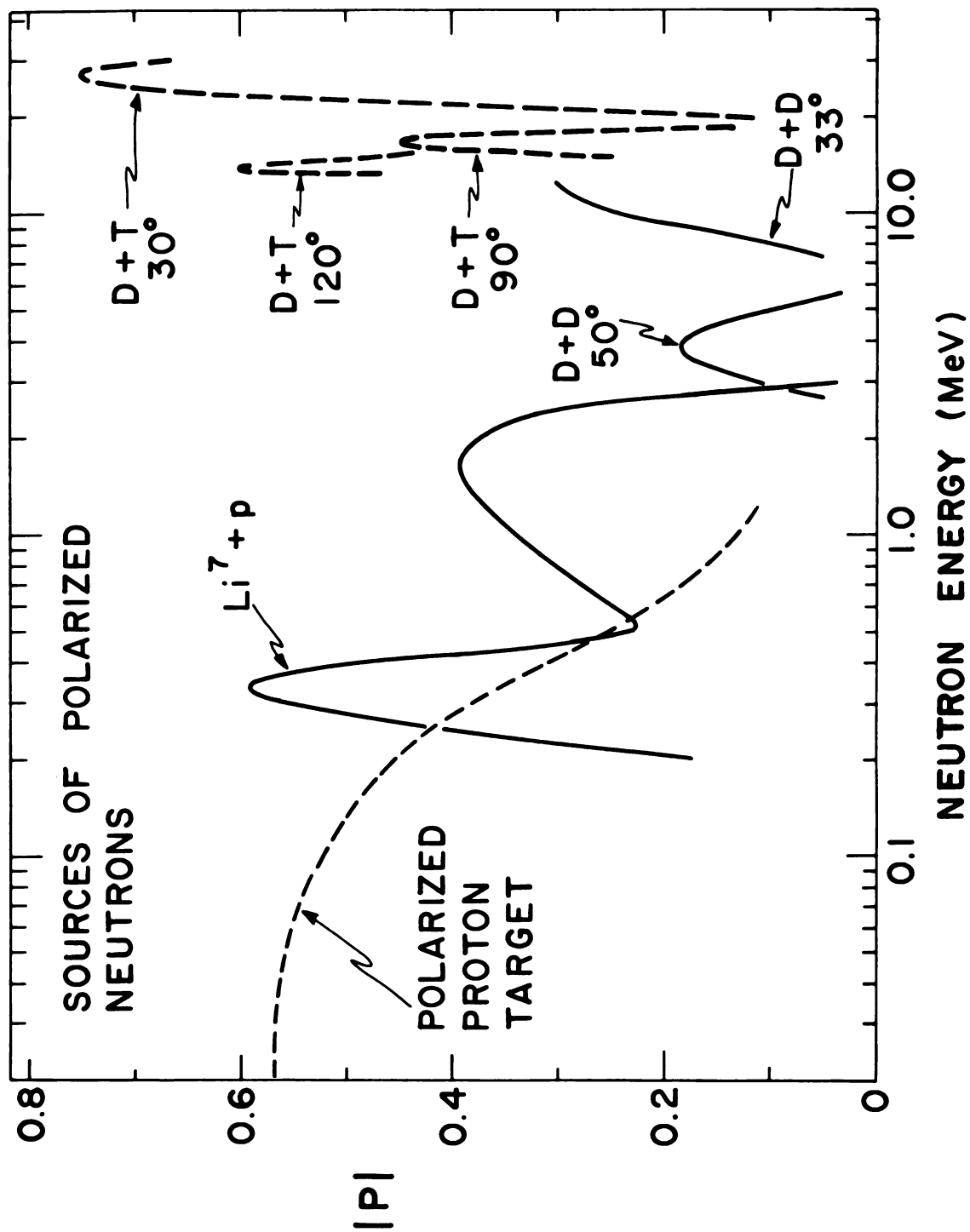


Fig. 12

TRIPLE - SCATTERING MEASUREMENT

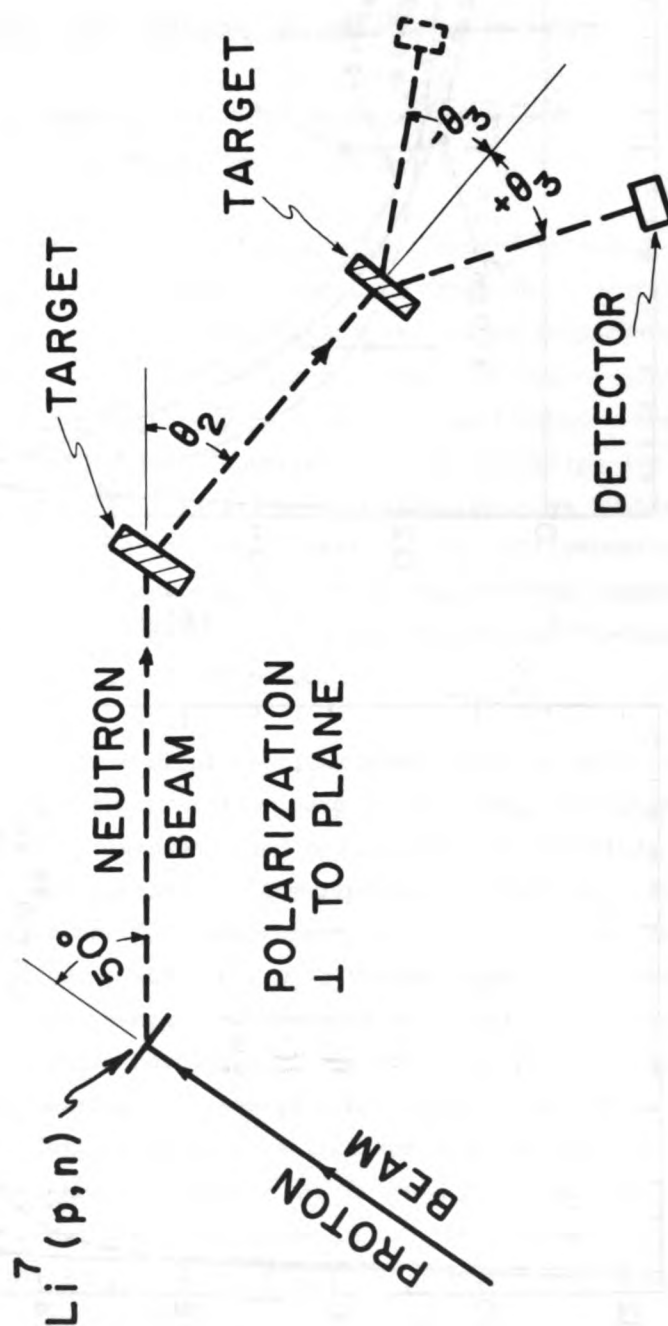


Fig. 13

SMALL-ANGLE SCATTERING ON U^{238}

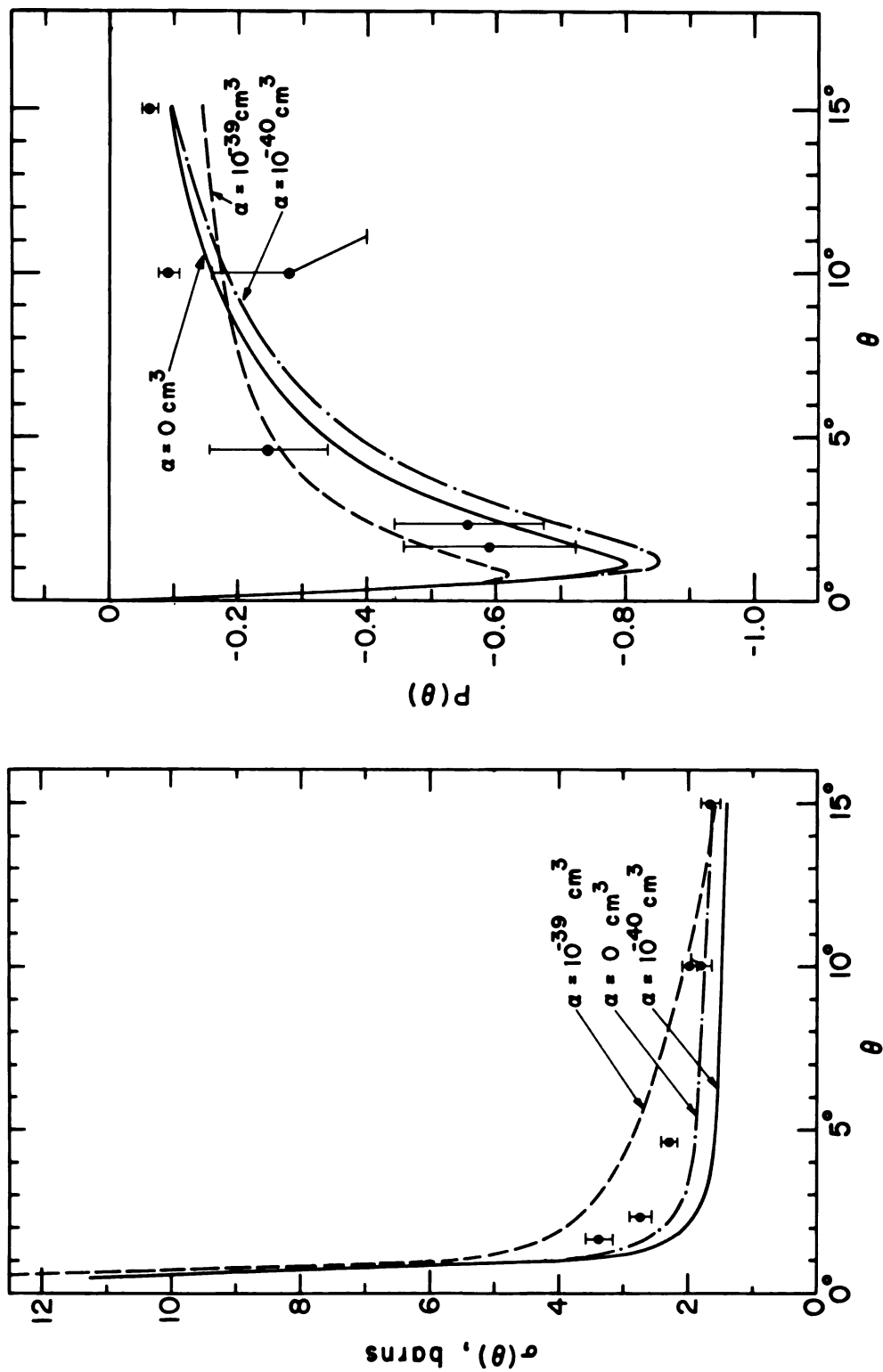


Fig. 14

NEUTRON CROSS SECTIONS AND RELATED MEASUREMENTS^{*}

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ABSTRACT: Except for the nuclear explosion with its extremely-high instantaneous intensity, fission sources are inferior to pulsed accelerator sources for high-resolution neutron cross-section measurements. A fission booster combined with a pulsed accelerator is a useful combination for cross-section measurements with low energy neutrons where moderate energy resolution is sufficient. Most of the neutron cross-section measurements requested for reactor design are possible with existing neutron sources. The greatest need for very intense neutron sources is for detailed partial cross-section measurements of the multiparameter type. Recent theoretical investigations suggest that highly-excited compound-nucleus states may have relatively simple configurations and are not "featureless" compound states beyond theoretical description.

1. INTRODUCTION

The interest in neutron cross sections comes from two groups: nuclear engineers who need neutron cross section data in the design and understanding of reactors, and nuclear physicists who need neutron data in the study of nuclear structure. If one surveys the latest "Compilations of EANDC Requests"⁽¹⁾ one sees that most of the requests are for measurements with resonance and fast neutrons and relatively few are for measurements with thermal energy neutrons. In fact most of the requests for thermal energy measurements are for condensed-matter inelastic-scattering measurements which have been discussed in the first paper of this session. Also thermal-neutron cross-section measurements are of little value to nuclear physics unless details of the reaction are also studied, such as the gamma-ray spectra from thermal-neutron capture or the mass distribution, etc. from thermal neutron fission. Since the other paper in this session and the discussion panels will consider these two fields, I will restrict my review to measurements made with resonance-energy and fast neutrons and consider only the need for higher fluxes for measurements with neutrons of these energies.

A. Requested Neutron Cross Sections

Most of the requested cross sections⁽¹⁾ are of the "one-parameter" type, $\sigma_T(E)$, $\sigma_\gamma(E)$, $\sigma_F(E)$, $\sigma_p(E)$, etc. For example, only the cross section for a

particular process such as neutron capture as a function of neutron energy is required and not the spectra of the gamma rays for each neutron energy. The exceptions to this statement are (1) elastic scattering cross sections where the angular distribution of the elastically scattered fast neutrons is required and (2) inelastic scattering cross sections where the energy distribution of the outgoing neutrons (or the gamma rays) is required and often the angular distribution of these inelastically scattered fast neutrons (or the gamma rays) is requested in addition. Since reactor calculations are made with a finite number of neutron energy groups, the cross sections in general are not needed with extremely good neutron energy resolution. However, Monte Carlo methods using detailed cross-section data will assume increasing importance in future reactor calculations. Most of the cross sections which are required are for isotopes which are available in reasonable quantities or for the natural element. In general, if a cross section is small, its value is not needed to high accuracy. Most of the desired cross-section measurements for the fissile and fertile materials as well as for many of the measurements on the moderator, structural and shielding materials are possible with existing* neutron sources. However, with the limited manpower and existing facilities, many, many years will be needed to satisfy all the requests. Although an increase in the strengths of neutron sources would increase the production rate some, other factors are much more important.

B. Neutron Cross Sections for Nuclear Structure

Although a great deal of information has been obtained concerning nuclear structure at high excitation from high-resolution "one-parameter" type experiments such as measurements of neutron total cross sections, it is now apparent that additional information from multi-parameter experiments is required. For example, the energy spectrum of the gamma rays from neutron capture in individual resonances, or the fission-product mass distribution from individual resonances will help determine the character of the capturing states. These two experiments can readily be expanded into multi-parameter experiments where coincidence and angular-correlation measurements of the capture gamma rays are made or the alpha, neutron or gamma-ray emission from fission is correlated to the mass distributions. These multi-parameter experiments will require very intense sources as will be discussed later.

2. DESIRABLE CHARACTERISTICS OF NEUTRON SOURCES FOR CROSS SECTION MEASUREMENTS

Neutron sources can be divided into two classes: monoenergetic neutron sources with a small energy spread ($\leq 1\%$) and polyergic sources with a large

* These include nuclear explosions and the ORNL electron linac under construction.

energy spread often extending over many decades of energy. The monoenergetic sources may be either steady-state or pulsed sources and much cross-section data have been obtained below 10 eV using a crystal spectrometer at a reactor and in the keV and MeV-energy regions using electrostatic accelerators. To use polyergic sources for high-resolution cross-section measurements, the neutrons must be produced in short time bursts and the time-of-flight technique must be used to determine their energies. Cross-section measurements of many different types have been made with pulsed neutron sources from thermal energies to ~ 10 MeV and for most measurements pulsed sources are superior to steady-state monoenergetic sources.

There are many factors which must be considered in evaluating the advantages of a particular neutron source for a specific cross-section measurement, such as intensity, neutron energy distribution in the source, energy resolution desired, background, instantaneous intensity, repetition rate for pulsed sources, etc. For any particular experiment there will be a compromise between intensity and resolution. The figure of merit of a pulsed source for high-resolution resonance-energy cross-section measurements (with energy resolution $\Delta E/E$) is approximately

$$\frac{(Pw)f}{w^2 + \tau^2}$$

where P is production rate during the burst of the useful neutrons within the energy range to be investigated

w is the duration of the burst including moderation time if required

(Pw) is the number of useful neutrons per burst

f is repetition rate of the bursts, and must be sufficiently low to prevent overlap

(Pw)f is the average production rate of useful neutrons

and τ is the time jitter of the detection equipment.

The magnitude of τ depends on the experiment but may be as large as a microsecond in the eV-energy region and as small as a nanosecond in the MeV-energy region.

Pulsed neutron sources have been made in a variety of ways: a fast chopper on a steady-state reactor, a pulsed reactor, a photoneutron target on an electron linac, a booster on a pulsed electron accelerator, a spallation target on a high-energy proton cyclotron, and a nuclear explosion. Excluding the nuclear explosion, existing pulsed accelerators are capable of producing approximately the same number of useful neutrons in considerably shorter pulses than can be obtained from existing fission neutron sources. Hence, considering the figure of merit only, pulsed accelerator sources are superior to pulsed fission sources (excluding a nuclear explosion) for high-resolution measurements in the keV and MeV-energy regions because of the short (\sim nanoseconds) pulses which can be obtained from the accelerator. Comparisons of the pulsed neutron sources will be made in much greater detail later

on during this seminar. However, I want to emphasize this conclusion now since most of the high-resolution measurements that I will discuss in my talk were made with high-intensity, short-pulse accelerators, and still greater intensity is required for many desirable experiments.

In considering the usefulness of a pulsed neutron source, there are several other factors in addition to the figure of merit defined earlier. The background, resulting from gamma rays and from undesirable energy neutrons produced in the source, can often render a particular experiment impractical even though there is a sufficient intensity of good neutrons. For some experiments it may be desirable to produce neutron bursts with an energy spread of a factor of only 10 or even less in order to reduce backgrounds from unwanted neutrons. This technique is often used with pulsed electrostatic accelerators; in addition the neutrons from the source are sometimes restricted in angle due to the kinematics of the reaction. If the background not associated with the source is large (such as in the measurement of the fission cross section of a highly alpha-active sample) it is desirable to have an extremely-high production rate during the burst such as is available with a nuclear explosion. For coincidence measurements the duty cycle should be high; hence, a source with a high repetition rate is desirable.

Some sources such as a fast chopper, crystal spectrometer and Van de Graaff require much smaller samples for transmission measurements than the moderated pulsed accelerator sources. Activation measurements have been made at low energies using a nuclear explosion; however, for higher energies (≥ 10 keV) monoenergetic sources are superior. It will also be difficult for pulsed sources with broad energy spreads to compete against monoenergetic (also pulsed) sources for high-resolution inelastic-scattering measurements. These are only a few of the more obvious factors which can determine the value of a neutron source for a particular cross-section experiment.

3. NEUTRON CROSS SECTIONS REQUESTED FOR REACTORS

The requests listed in the latest compilation are approximately equally divided between the requests for the fissile and fertile materials and the requests for moderator, structural and shielding materials. The problem for the fissile nuclides is that many of the cross sections are needed to high accuracy over a wide energy range such as the fission cross section of ^{235}U requested to an accuracy of 1 - 2% over the energy range from thermal to 8 MeV. At the present time after many, many man years of effort there are several $\pm 3\%$ measurements which disagree with each other by as much as 7%. At a recent conference on Neutron Cross Section Technology, it was estimated⁽²⁾ that to measure the (n,α) cross section of ^6Li to a few percent accuracy over a wide energy range has required ~ 30 scientific man years of effort at an estimated cost of a million dollars. For most of the measurements on the fissile and fertile materials,

existing high-intensity sources are suitable, with one notable exception. In order to determine the Doppler reactivity coefficient for a fast reactor the detailed resonance structure of ^{238}U and ^{239}Pu must be measured with extremely good energy resolution ($\Delta E/E \approx 10^{-4}$) from ~ 10 to ~ 100 keV. This resolution is nearly an order of magnitude better than can be obtained at present and, hence, requires a much more intense source.

For the moderator, structural and shielding materials, the two most requested types of cross sections are the capture cross sections up to a few hundred keV and the inelastic-scattering cross sections up to ~ 10 MeV. Capture cross sections for materials like Na, Cr, Fe, Ni, Mo, etc. are requested to an accuracy of $\sim 10\%$ or ~ 5 mb. Capture measurements have been made for years with various kinds of detectors with both monoenergetic and pulsed neutron sources. Slide 1 shows some recent results on Na obtained in a 6-hour run by Block and collaborators⁽³⁾ using the RPI linac and a large scintillation tank for detecting the gamma-rays. A few of the observed resonances such as those at 53.4, 240, and 300 keV were known from transmission measurements and have neutron widths of several keV. These data have not yet been analyzed but assuming that the radiation widths of the large resonances are probably comparable to the value of 0.6 eV obtained for the 2.85-keV resonance, the ratio of capture to scattering in these large resonances is $\sim 1:1000$. The efficiency for detecting scattered neutrons with the scintillation tank has been reduced to less than 10^{-5} that for detecting neutron capture; hence, very small capture cross sections (\sim mbs) can be easily measured. Capture results have also been obtained by other techniques, i.e., using Moxon-Rae detectors at the Harwell linac, with a pulsed Van de Graaff at ORNL, and by groups at LASL using a nuclear explosion, etc. I conclude that existing neutron sources are sufficiently intense for the capture cross-section measurements needed for reactors.

Inelastic cross sections for materials like Na, Si, Ca, Fe, Pb, etc. are requested to an accuracy of $\sim 10\%$ up to ~ 10 MeV. The most suitable source for these measurements is a pulsed (\sim one-nanosecond) monoenergetic source from an electrostatic generator. Slide 2 shows some recent data on Fe⁽⁴⁾ obtained in 80 minutes with 5-MeV neutrons from a (d,d) source using a pulsed Van de Graaff at ORNL. The energy resolution of the elastically scattered neutrons is determined principally from the time-of-flight resolution (2-nanosecond bursts at 4 meters) and that of the low-energy inelastic groups are determined principally from the energy spread of the incident neutrons (~ 60 keV). Since measurements must be made for many incident neutron energies (and often different angles of scattering, for which multiple detectors can be used), a complete set of measurements can be very time consuming. An increase in the source strength would be very desirable for

inelastic scattering measurements above a few MeV. Below ~ 2 MeV the present sources are sufficient for inelastic scattering as demonstrated by the vast amount of data from A. B. Smith and collaborators at ANL.

An alternate approach to obtain the integrated (over angle) inelastic-scattering cross section is to measure the energy spectrum of the emitted gamma rays as a function of neutron energy. Most of the measurements of this type have been made using monoenergetic sources, but a polyergic source and the time-of-flight technique have also been used. Data have been obtained with both NaI and Li-drifted Ge detectors. For poor-resolution measurements (which would probably satisfy many of the reactor requests) it is possible that a proton recoil detector could be used to measure the spectra of the scattered neutrons using a pulsed polyergic source; hence, all inelastic spectra data for all incident neutron energies could be obtained simultaneously. A high-intensity, short-pulse source such as the ORNL electron linac should have sufficient intensity for this proposed technique. It seems very unlikely that double time-of-flight measurements would be possible.

4. NUCLEAR STRUCTURE OF HIGHLY EXCITED STATES

A. Nuclear Models for Highly Excited States

Soon after several very narrow resonances (~ 0.1 eV) were observed with low energy neutrons a few decades ago, they were explained on the basis of the compound nucleus model. In the compound nucleus model the excitation energy is shared amongst all the nucleons in the nucleus. Since the structure is so complicated it takes a great many oscillations for the nucleus to approach a configuration where it can emit a neutron, or a gamma, or undergo fission, etc., which accounts for the long lifetimes and, hence, the narrow widths observed for low-energy resonances. Some 15 years ago a gross structure (width of several MeV) was observed in neutron cross sections, and these were soon explained from an optical nuclear model as resonances in a complex nuclear potential well. The incoming neutron acts as a single particle in the field created by the other nucleons in the nucleus. This optical model is a natural continuation of the shell model of the nucleus (which has been so successful in understanding the low-lying bound excited states) into the unbound region where neutron emission is possible. Hundreds of theoretical papers have been published on the compound and optical nuclear models and there is no question of the usefulness and accuracy of these models, in spite of their apparent contradiction. In the past few years experimental data have suggested cross section variations which are more rapid than those explained from the optical model and yet much wider than compound-nucleus resonances. The experimental data to determine this intermediate structure needs to be very comprehensive and accurate. The greatest need for more intense fluxes is for these measurements.

With the present day high-resolution neutron spectrometers, often over a hundred "compound-nucleus" resonances can be observed in a single isotope. The parameters of the resonances have been analyzed to investigate distributions of the parameters, such as the spacing distributions and the width distributions, and there is good agreement between experiment and theory. Possibly more experimental data would produce discrepancies which could be significant. By averaging over these "fine-structure" resonances or by appropriate poor-resolution measurements the quantity (Γ_n^l/D) , known as the strength function for s-wave or p-wave neutrons, can be obtained. The experimental data are in fair agreement with an appropriate spheroidal optical model which has a diffuse surface, surface absorption, spin-orbit coupling, etc. However, the experimental points do scatter more than would be expected from their quoted errors. Also some experimental data seem to indicate there are local variations of the strength function, i.e., that the s-wave strength function varies over an energy region of $\sim 50 \text{ keV}^{(5)}$ or even \sim hundreds of eV.⁽⁶⁾ More extensive data are needed to investigate this effect for many nuclides. Also in the study of the spacings of nuclear levels by the method due to Dyson,⁽⁷⁾ there has been the hint that there might be long-range correlations among "fine-structure" resonances which would be very surprising. However, the fact that very small resonances might be missed and that small resonances which may be p-wave resonances are included in the consideration make the present evidence rather untrustworthy. High-resolution capture measurements to detect the very small resonances and measurements to determine their l -values are required. These measurements will probably require more intense sources than we have today.

B. Intermediate Structure from Doorway States

In order to understand the intermediate structure observed recently in neutron cross section data, Feshbach⁽⁸⁾ and co-workers have suggested a mechanism called a "doorway" state. A doorway state is simply the state that a reaction must go through as the first stage towards forming the compound nucleus. Doorway states involve comparatively-simple excitations and their nature depends on the nuclear model. If one considers the shell model, the doorway state for an even-even target nucleus can consist of only two-particle one-hole states since this is the only kind of state which a two-body residual-interaction operating on the target nucleus plus incident particle will generate. In this model the doorway states are called 3-quasi-particle states (2 particles + 1 hole). These three-quasi-particle states can change into more complex states such as 5-quasi-particle states, then 7, etc. via successive two-body interactions until a "true" compound nucleus is formed. If one considers the collective model, the doorway state might consist of the target nucleus excited to a collective state (such as the dipole or quadrupole vibrational state) with the incident particle dropping into a single-

particle level. Other possibilities have also been suggested. In order to obtain experimental data concerning these doorway states one can either make poor-resolution measurements and average over the compound-nucleus resonances experimentally or make high-resolution measurements over a broad energy range (hundreds of keV) and then average the neutron strengths over the resonances numerically. If the experimental resonances are sufficiently widely spaced as in the case of ^{208}Pb , one can compare the actual number of predicted 3-quasi-particle states and their neutron strengths to the experimental data.

C. Three-Quasi-Particle "Doorway" States

Using this "doorway" state approach, Shakin⁽⁸⁾ has calculated the 2-particle 1-hole states with spin $1/2$ and even parity which might be produced by neutrons upon the even isotopes of Pb and Sn. Theory predicts seven s-wave resonances in ^{208}Pb with neutrons up to 2 MeV and the sum of their neutron widths was predicted to be ~ 50 keV. Experimentally only 2 or possibly 3 resonances have been observed and their neutron widths add up to ~ 120 keV. Figure 3 shows a few angular distributions⁽⁹⁾ (obtained with a neutron energy resolution of a few keV) which are necessary in order to interpret the total cross section shown in the figure. The s-wave resonance in this energy region occurs at 1.735 MeV and has a width of 60 keV. There are many odd-parity resonances, including four f-wave resonances which have rather large neutron widths (a few percent of the Wigner limit, which is the limit for a single-particle resonance) which are possible candidates for doorway states. In addition to 2-particle 1-hole doorway states, odd-parity states could also be produced by coupling the positive-parity neutron states to the 3^- collective state of ^{208}Pb . More-intense neutron sources would lower the limit for the size of the small resonances which might be missed as well as make it possible to extend the measurements to higher energies.

Shakin's calculations for the even isotopes of tin predicted many 2-particle 1-hole states with a few hundred keV spacing and an s-wave strength function which decreased from $(\sim 0.7 \text{ to } \sim 0.1) \times 10^{-4}$ for the even tin isotopes. The experimental s-wave strength function determined from transmission measurements⁽¹⁰⁾ up to a few keV is in good agreement with this theory; however, more experimental data are desired from higher energy measurements. The experimental level spacings are a few hundred eV indicating that the strengths of the doorway states have been spread over ~ 1000 compound nucleus states. Also several strong p-wave resonances were observed which might be due to the effects of doorway states.

There are also other measurements of partial cross sections such as (n,α) and (n,p) which indicate the existence of intermediate structure; however, more and higher statistical accuracy data are needed. Feshbach⁽⁸⁾ has emphasized that it

is essential to minimize the number of channels for the nuclear reaction in order to enhance the fluctuations. One should therefore examine partial cross-sections such as σ_γ , σ_p , σ_α (and also the spectra of the emitted particles) in order to specify as many quantum numbers of the emergent particles as possible. To detect the intermediate structure, these cross sections must be measured over a wide energy range and if possible with good energy resolution to see how the doorway states are distributed to the compound-nucleus states.

D. Analogue States

Perhaps the clearest example of a doorway state is the analogue state produced from an isobaric analogue reaction. Figure 4 shows some data on the $^{40}\text{Ar}(p,p)$ reaction. The top curve shows some early, poor-resolution data and two apparent resonances are observed at 1.87 and 2.45 MeV whose spins and parities were determined to be $3/2^-$ and $1/2^+$. These two resonances correspond to the analogue states of the 4th and 6th excited states of ^{41}Ar whose spins and parities are $3/2^-$ and $1/2^+$. The middle curve of this figure shows recent high-resolution data⁽¹¹⁾ from Duke University for the first third of the upper curve. The proton energy spread in this experiment was only 200 eV; hence, $\Delta E/E$ is only 10^{-4} . There are many resonances (one as narrow as 120 eV) in the region near 1.87 MeV. The bottom curve covers a 60 keV interval and all of these resonances correspond to $3/2^-$ states. Some twenty strong $3/2^-$ resonances were found spread over an energy region with a width of ~ 100 keV, whereas the $3/2^-$ resonances outside this region were much weaker. This is a beautiful example of a doorway state some 60 keV wide whose strength gets distributed amongst the fine-structure states of the compound nucleus.

Figure 5 shows high-resolution data in the region of 2.45 MeV, the region of the expected 6th analogue state ($1/2^+$). The large resonances in the region around 2.45 MeV have spin $1/2^+$. Most of the proton strength of the $1/2^+$ resonances is located in a ~ 100 keV interval. This is another example of a doorway state. Two other proton-induced reactions were investigated in this work, namely the (p,α) and the (p,n) reactions. The (p,α) reaction does show some effect of the 1.87-MeV analogue resonance, but none in the 2.45-MeV resonance. The (p,n) reaction which is energetically possible only for the 2.45-MeV resonance did not show any effect. This can readily be understood since the spin of ^{40}K is 4, which would require f-wave emitted neutrons from these $1/2^+$ states whose penetrability would be small. Hence, if one were to measure the $^{40}\text{K}(n,p)$ reaction, this analogue state would not enhance the cross section. It is expected that (n,p) reactions will not produce analogue states due to the excess of neutrons in all but the lightest nuclides; however, it is possible that intermediate structure would appear from other doorway states.

E. Doorway States from Neutron Capture

It has also been suggested that doorway states can be determined from the capture of either low-energy or fast neutrons. In comparing thermal neutron capture gamma-rays to (d,p) reaction yields for ^{54}Fe and ^{56}Fe , Ikegami and Emery⁽¹²⁾ observe an anti-correlation between the (n, γ) and (d,p) reaction, i.e., the gamma-ray transitions go to regions of excitation of the compound nucleus where the single-particle stripping strength is small. This is shown for ^{56}Fe in Figure 6 where the spectroscopic factor for the (d,p) reaction is compared to the dipole strength for the gamma-ray transition. Comparisons between the high-energy gamma-rays from neutron capture and the (d,p) process have been made throughout the periodic table by Groshev.⁽¹³⁾ The strengths of the strong E1 gamma rays to low-lying p-states are in agreement with the large spectroscopic factors for $l = 1$ groups from the (d,p) reaction. This agreement has been taken as evidence for the direct capture process. This process follows from the optical model, since the neutron can radiate when it is in a region of changing nuclear potential. However, in the case of ^{54}Fe and ^{56}Fe , Ikegami and Emery suggest that the anti-correlation arises from 3-quasi-particle (seniority 3) states which radiate to low-lying states which are not simple seniority-one states fed in the (d,p) reaction. For ^{55}Fe , for example, the 2-proton 1-neutron state $[(f_{7/2})_p^{-1} (f_{5/2})_p (d_{5/2})_n]_{1/2+}$ is expected near the neutron binding energy; this state has allowed E1 transitions to the states $[(f_{7/2})_p^{-1} (f_{5/2})_p (p_{3/2})_n]_{1/2-, 3/2-}$ which are expected at ~ 4 MeV. Similar configurations involving two neutrons and a neutron hole are also available. Ikegami has also pointed out that some of the seniority-three excitations near the neutron binding energy have allowed E1 transitions to seniority-one (single-neutron) configurations. For example, the three-neutron configurations $[(f_{7/2})_p^{-1} (g_{9/2}) (p_{3/2} \text{ or } p_{1/2})_n]_{1/2+}$ have allowed E1 transitions to the single-neutron $p_{3/2}$ and $p_{1/2}$ configurations, respectively. This could be the explanation for the strong excitation of both the ground state and the 14-keV excited state in ^{57}Fe by the (n, γ) reaction. The ground state is known to be predominantly seniority 3, $[(f_{7/2})_p^{-2} + (p_{3/2})_n]$, while the 14-keV excited state is of seniority 1, $(p_{3/2})$.

The existence of three-quasi-particle doorway states has been introduced to explain strong high-energy M1 transitions⁽¹⁴⁾ from neutron capture in a few s-wave resonances in the even-isotopes of tin. In Shakin's calculations for the neutron widths of the 3-quasi-particle neutron states of the isotopes of tin, two states of the following configurations were quite strong:

$[(g_{9/2})_p^{-1} (g_{7/2}) (s_{1/2} \text{ or } d_{3/2})_n]_{1/2+}$. These 2 states could decay by M1 transitions to the low-lying $s_{1/2}$ and $d_{3/2}$ single-neutron states respectively. However, for lack of intensity from a fast chopper, these NaI gamma-ray measurements were only possible for neutron resonances below 400 eV; with more intense neutron

sources it would be very desirable to extend the measurements to higher energies and use Li-drifted Ge detectors.

In a recent paper⁽¹⁵⁾ on the energy levels of ^{170}Tm from (d,p) and thermal-neutron capture gamma-ray measurements, an interesting comparison has been made of the relative intensities of the 2 processes (Figure 7). The high-energy gamma rays populate the same states as the (d,p) process, which yields only neutron states. Since these high-energy gamma rays do not populate excited-proton states, the authors suggest that these high-energy gamma-rays arise from direct capture which utilizes only a small part of the thermal-capture cross section of 127 barns. However, the predicted direct capture cross section for this mass region is only 0.5 barn, which is quite small. Lane⁽¹⁶⁾ has computed the contribution from collective capture; however, it is only the same order as the direct capture. The $\text{Tm}(n,\gamma)$ data showed a very weak E1 transition to the 1^- ground state in contrast to the strong ground-state intensity from the (d,p) reaction. The authors⁽¹⁵⁾ suggest "this unusual forbiddenness is probably to be understood in terms of the specific character of the capturing state." However, I think that this emphasizes the need for gamma-ray spectra measurements in many resonances to allow for the Porter-Thomas variation in partial radiation widths since the thermal cross section of Tm is dominated by the strong 1^+ resonance at 3.92 eV.

F. Capture Gamma-Ray Spectra Measurements in Individual Resonances

To emphasize the value of gamma-ray spectra measurements from individual resonances, I would like to present some thermal-neutron capture gamma-ray measurements upon ^{182}W , ^{184}W , ^{186}W . In all 3 isotopes strong E1 transitions were observed to the $[510]1/2^-$ and $[510]3/2^-$ rotational states and weak transitions to the $[512]3/2^-$ state. It was considered that collective capture would be needed to explain the anomaly. However, capture spectra measurements from resonance capture have shown that although the transition to this $[512]3/2^-$ is weak in the 4.16-eV resonance (which dominates the thermal capture) this transition is strong in two other resonances. These capture spectra measurements⁽¹⁷⁾ upon W were made with a 20-cm³ Ge detector using the RPI linac. Figure 8 shows preliminary data for 3 resonances in ^{182}W . The 2 high-energy gamma-ray lines from the 4.1-eV resonance are E1 transitions to the $[510]1/2^-$ and $[510]3/2^-$ rotational states and no transition to the $[512]3/2^-$ state at 209 keV is observed. However, in the other 2 resonances the transitions to this state are strong.

Figure 9 shows data⁽¹⁸⁾ from Saclay for 4 resonances in ^{195}Pt obtained with a 6 cm³ Li-drifted-Ge detector 30 meters from the neutron source. The strong fluctuations in the intensities of the 3 high-energy gamma rays for these 4 resonances is striking. From spectra measurements from 22 resonances with spin and parity 1^- , the authors conclude that

the distribution of partial radiation widths is

in good agreement with a Porter-Thomas distribution.

Capture spectra measurements⁽¹⁹⁾ have been made in the keV energy region from resonances in ^{206}Pb using a large NaI crystal and a pulsed Van de Graaff at ORNL (Figure 10). For intensity reasons the neutron energy resolution was poor but sufficient to permit the measurement of the gamma-ray spectra from 5 resonances up to 75 keV. The resolution of the NaI crystal was sufficient to resolve transitions to the ground state and first 2 excited states of ^{207}Pb , which all have negative parity. Since 3 of the resonances are p-wave resonances, these high energy transitions must be M1. The authors⁽¹⁹⁾ have suggested that two-particle one-hole states based on the $1_{13/2} \rightarrow 1_{11/2}$ excitation for neutrons and $h_{11/2} \rightarrow h_{9/2}$ for protons are possibilities to explain these high-energy M1 transitions. To improve both the neutron energy resolution and the gamma-ray energy resolution by using Ge-detectors would require a considerable increase in the intensity of the neutron source which is expected from the ORNL linac.

Sometimes in thermal-neutron capture gamma-ray measurements there is an ambiguity whether a gamma ray is from the capturing state or a cascading gamma ray. From capture spectra measurements in the keV energy region this ambiguity can often be resolved since the gamma ray energy will be increased by the energy of the neutron if the gamma-ray is a primary transition.

G. Other Neutron Cross Section Measurements Requiring Intense Neutron Sources

There are many other high-resolution, multiparameter experiments of interest which require intense neutron sources; some will probably require more intense sources than are presently available.

(1) Gamma-ray spectra measurements from neutron capture between compound-nucleus resonances to investigate direct and semi-direct capture.

(2) Gamma-ray spectra from neutron capture in resonances of the fissile nuclides.

(3) Coincidence and angular-correlation measurements between cascading gamma rays (or between gamma-rays and conversion electrons) from neutron capture in individual resonances.

(4) A study of the gamma rays in coincidence with fission to determine the importance of the $(n,\gamma f)$ process.

(5) A survey of subthreshold fission for many resonances in the even-even heavy nuclides.

(6) Measurements of the energy spectra of alpha particles and protons from individual resonances and over a wide neutron energy range.

(7) Transmission measurements of very small samples, for example, using polarized targets and polarized neutrons to determine the spin state of individual resonances.

(8) Detailed studies of the fission process in individual resonances.

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FIGURE CAPTIONS

- Fig. 1. Capture cross section of Na in the keV energy region with an energy resolution of 2.5 nsec/m.
- Fig. 2. Inelastic scattering cross section of Fe at 5 MeV. The numbers above each peak are Q values in MeV.
- Fig. 3. Differential elastic scattering of ^{208}Pb with an energy resolution of a few keV.
- Fig. 4. Proton yield from $^{40}\text{Ar}(p,p)$ in the region of the analogue state of the 4th excited state of ^{41}Ar .
- Fig. 5. Proton elastic scattering cross section of $^{40}\text{Ar}(p,p)$ in the region of the analogue state of the 6th excited state of ^{41}Ar .
- Fig. 6. Comparison of the (n,γ) and (d,p) strengths for final states in ^{57}Fe .
- Fig. 7. Histogram showing the comparison of the energies and intensities of the (d,p) and (n,γ) spectra for thulium.
- Fig. 8. Gamma ray spectra from 3 resonances in ^{182}W using a 20 cm^3 Ge detector. The 2 strong gamma rays at 5.3 MeV in the middle curve are due to capture in the 18.8-eV resonance in ^{186}W .
- Fig. 9. Gamma ray spectra from 4 resonances in ^{195}Pt . The positions of the five high energy transitions are indicated by arrows.
- Fig. 10. Yield of gamma rays as a function of neutron energy for ^{206}Pb target.

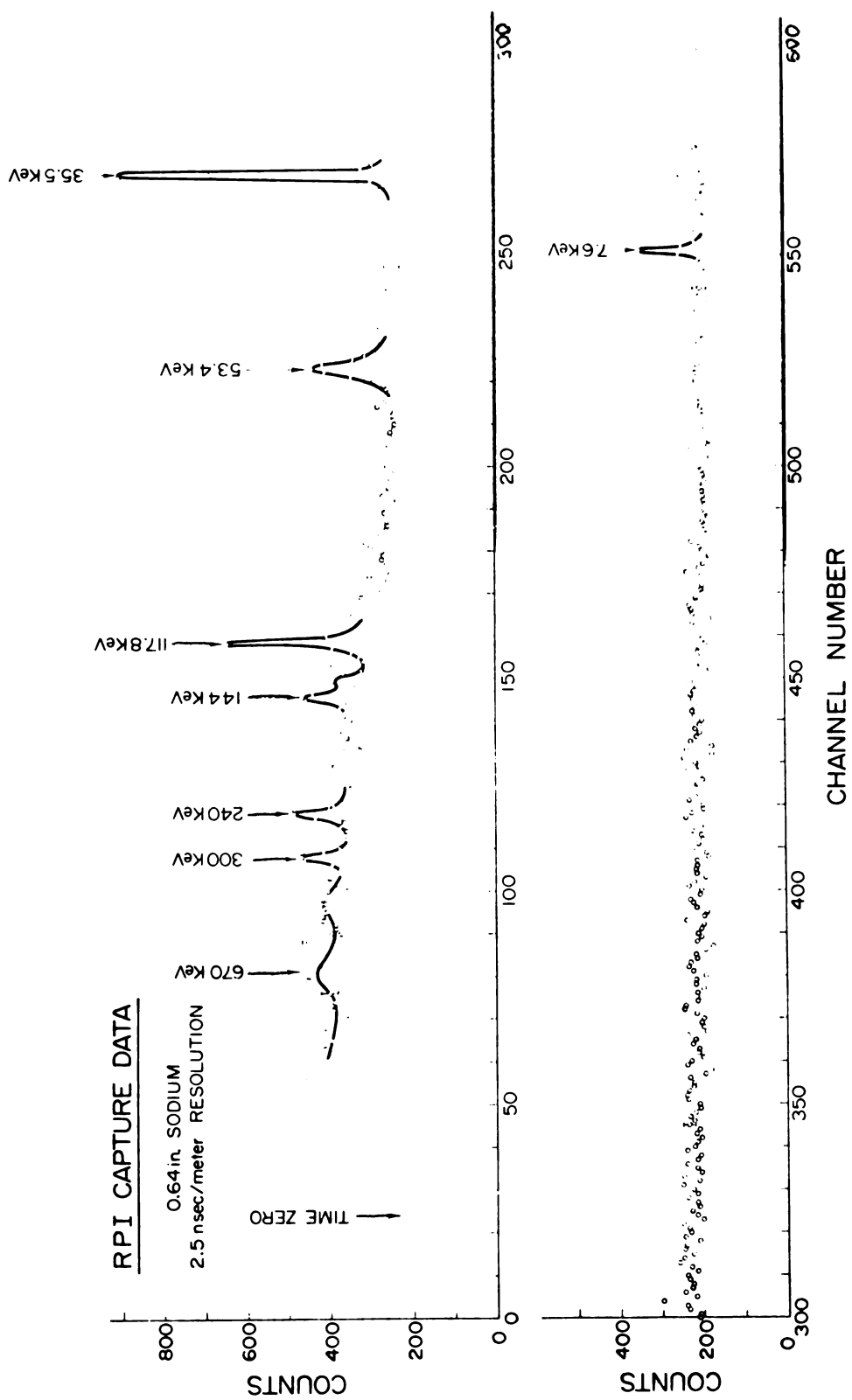


Figure 1

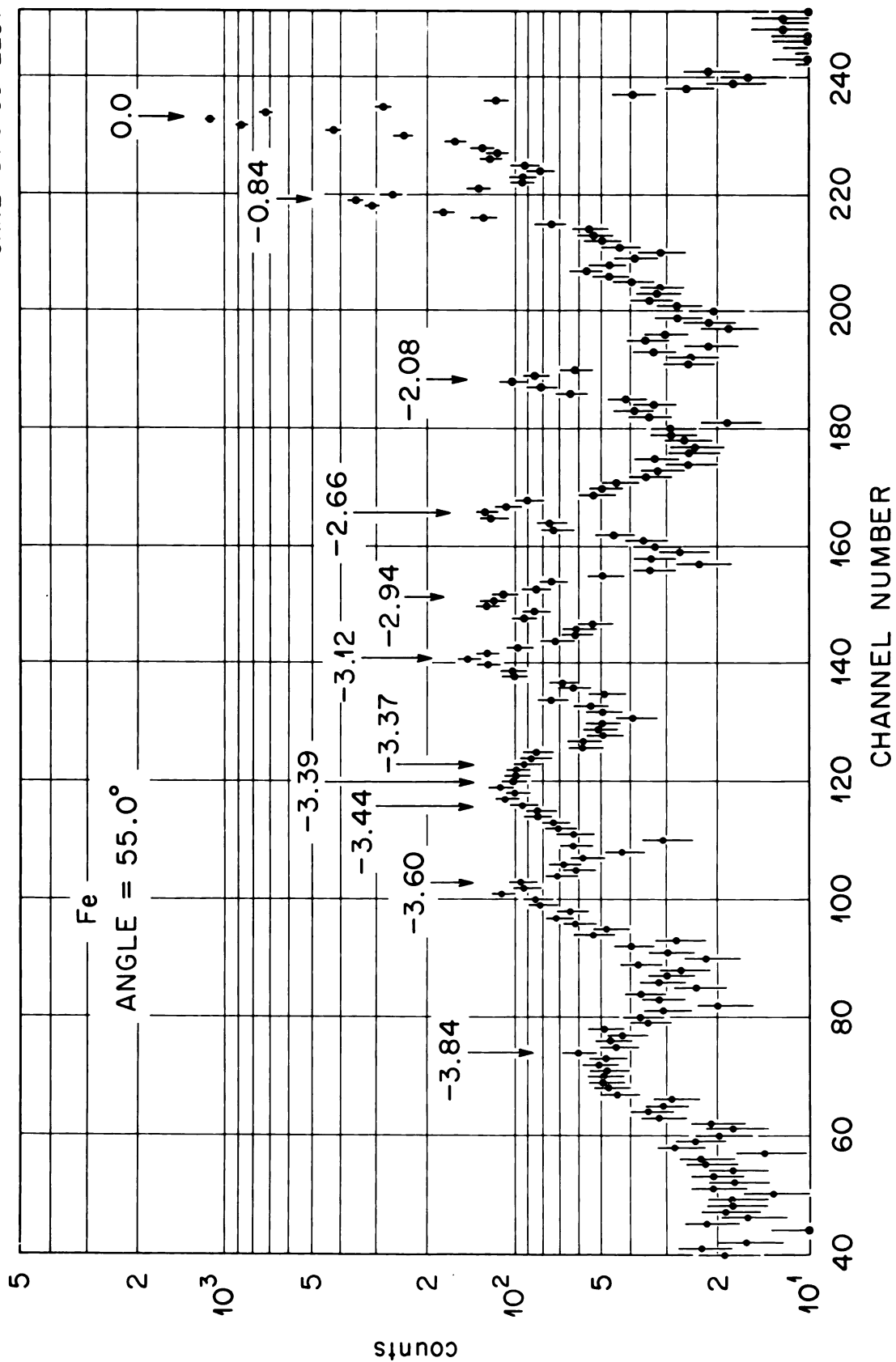


Figure 2

$$\begin{aligned}
 E_n &= 1.761 \pm 0.0020 \text{ MeV} \\
 \delta_s \frac{1}{2} &= +150^\circ \\
 \delta_p \frac{1}{2} &= -43^\circ; \delta_p \frac{3}{2} = -34^\circ \\
 \delta_d \frac{3}{2} &= +9^\circ; \delta_d \frac{5}{2} = -45^\circ \\
 \delta_f \frac{3}{2} &= -9^\circ \\
 \delta_f \frac{7}{2} &\begin{cases} \Gamma_2 = 0.0022 \\ \text{POT} = -24^\circ \\ E_0 = 1.7600 \\ \chi = 0.5 \end{cases}
 \end{aligned}$$

$$\begin{aligned}
 E_n &= 1.749 \pm 0.0020 \text{ MeV} \\
 \delta_s \frac{1}{2} &= +79^\circ \\
 \delta_p \frac{1}{2} &= -35^\circ; \delta_p \frac{3}{2} = -14^\circ \\
 \delta_d \frac{3}{2} &= -17^\circ; \delta_d \frac{5}{2} = -45^\circ \\
 \delta_f \frac{3}{2} &= -9^\circ \\
 \delta_f \frac{7}{2} &\begin{cases} \Gamma_2 = 0.0053 \\ \text{POT} = +10^\circ \\ E_0 = 1.7499 \\ \chi = 1.6 \end{cases}
 \end{aligned}$$

$$\begin{aligned}
 E_n &= 1.701 \pm 0.0020 \text{ MeV} \\
 \delta_s \frac{1}{2} &= +71^\circ \\
 \delta_p \frac{1}{2} &= +5^\circ; \delta_p \frac{3}{2} = -33^\circ \\
 \delta_d \frac{3}{2} &= -63^\circ; \delta_d \frac{5}{2} = -12^\circ \\
 \delta_f \frac{5}{2} &\begin{cases} \Gamma_2 = 0.0030 \\ \text{POT} = 0^\circ \\ E_0 = 1.7010 \end{cases} \\
 \delta_f \frac{7}{2} &= +8^\circ \\
 \chi &= 1.7
 \end{aligned}$$

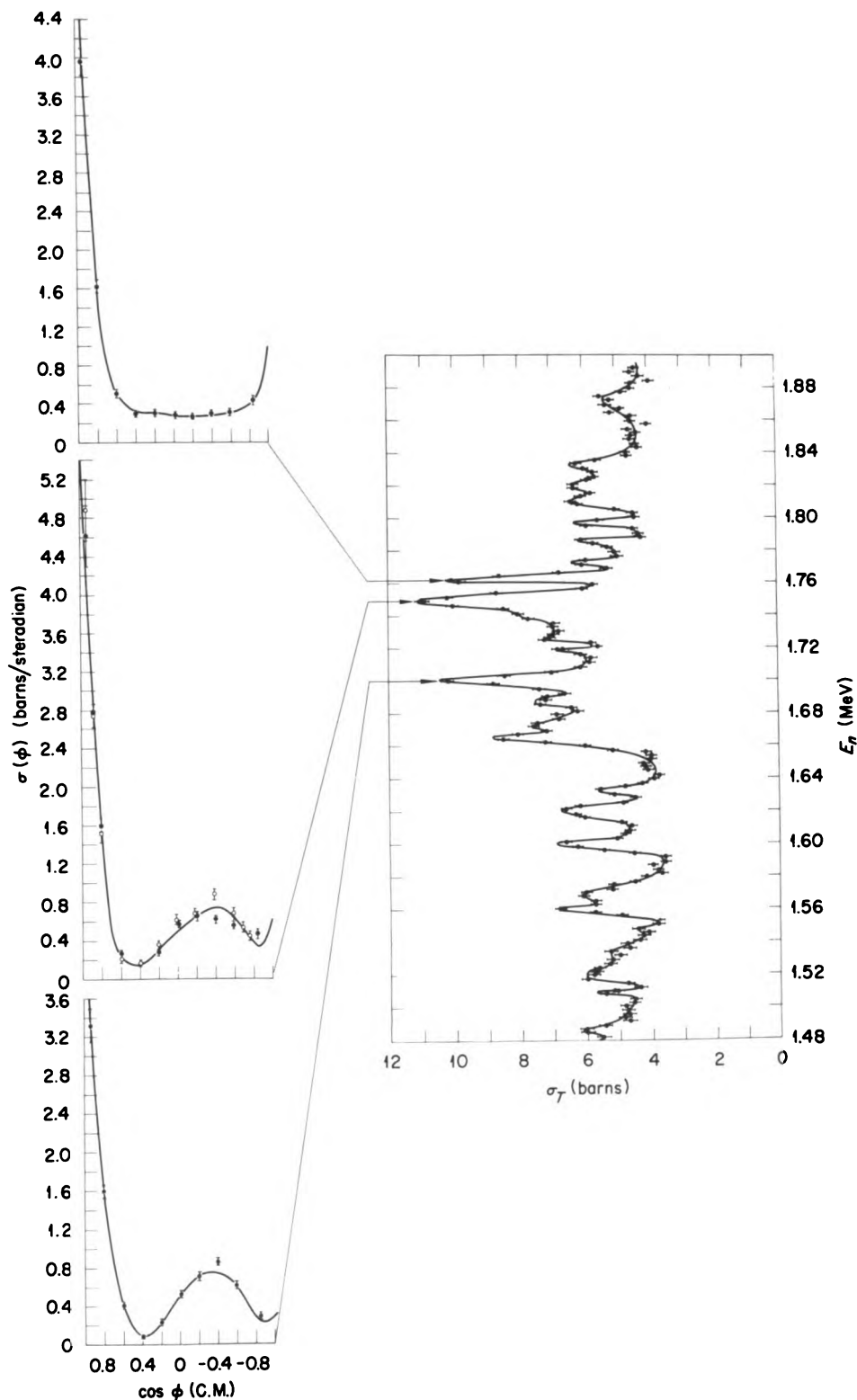


Figure 3

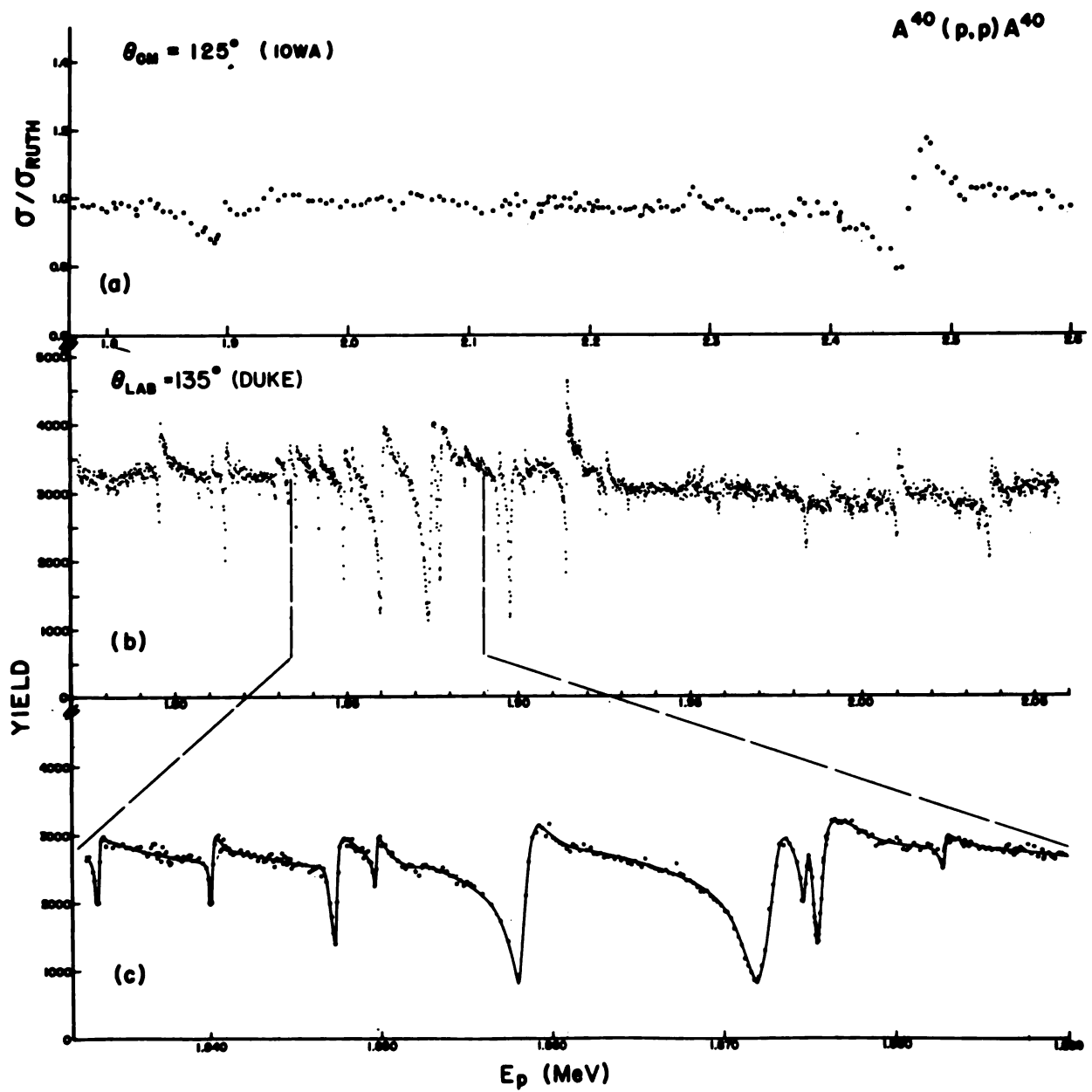


Figure 4

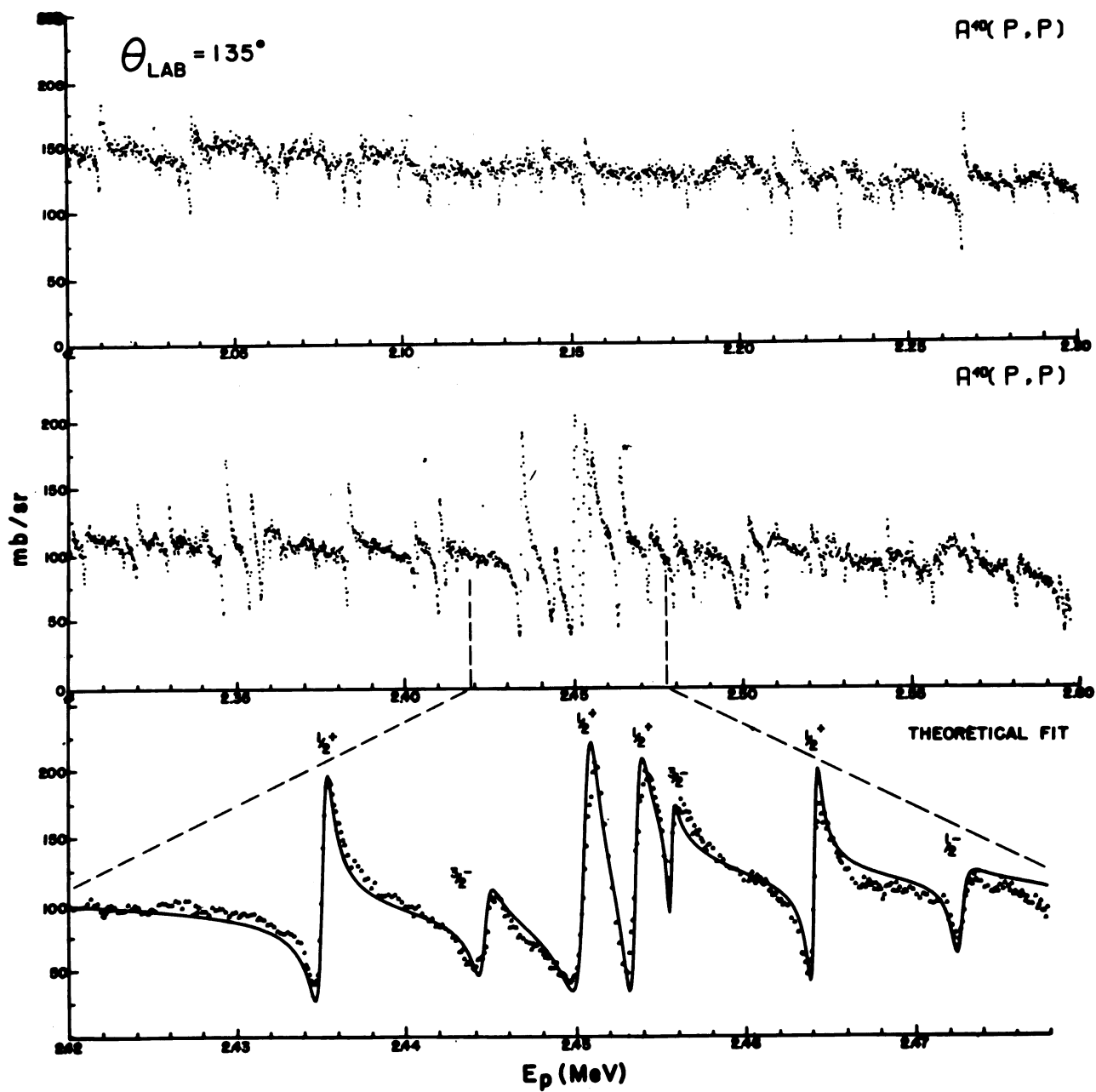


Figure 5

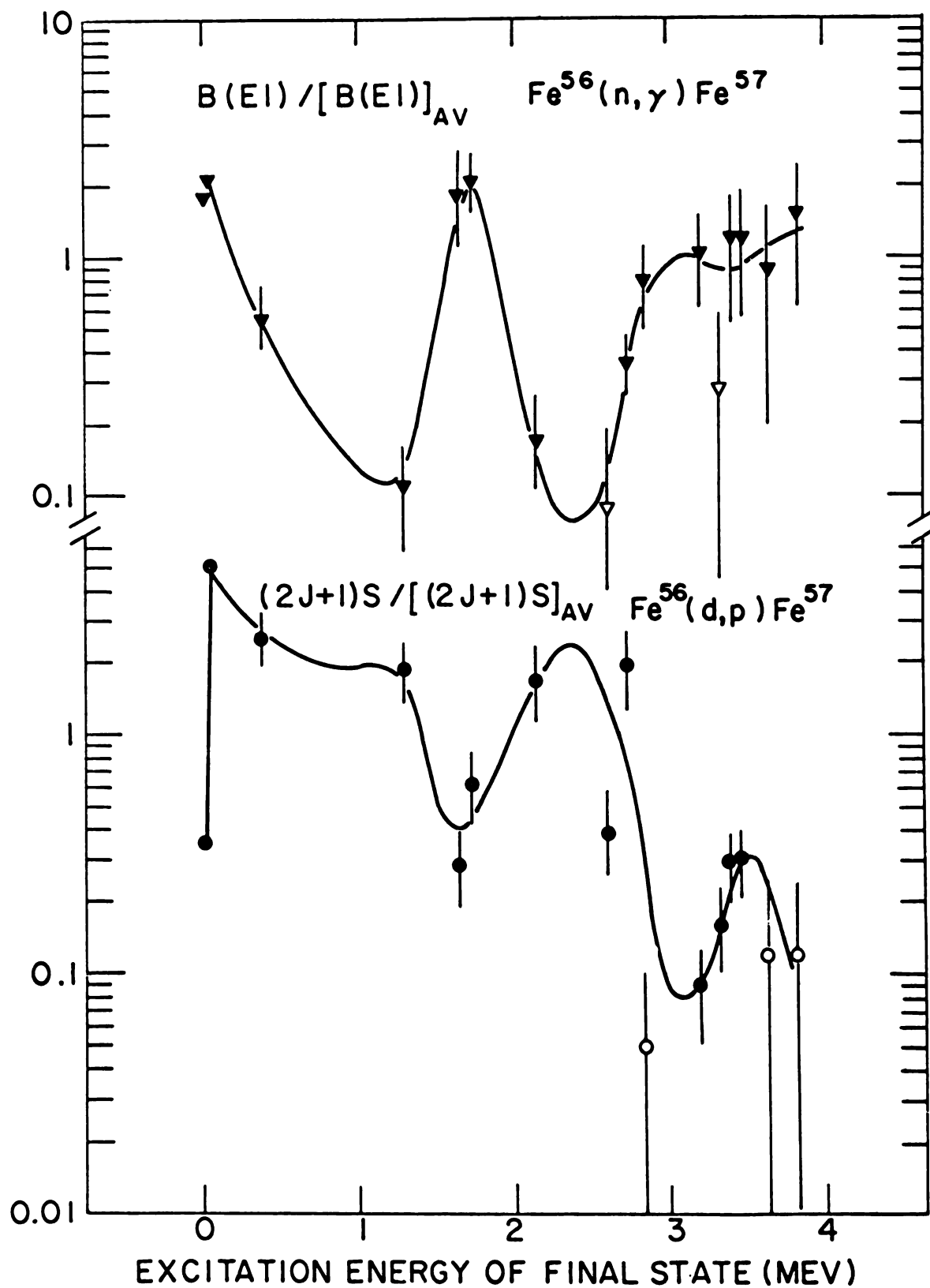


Figure 6

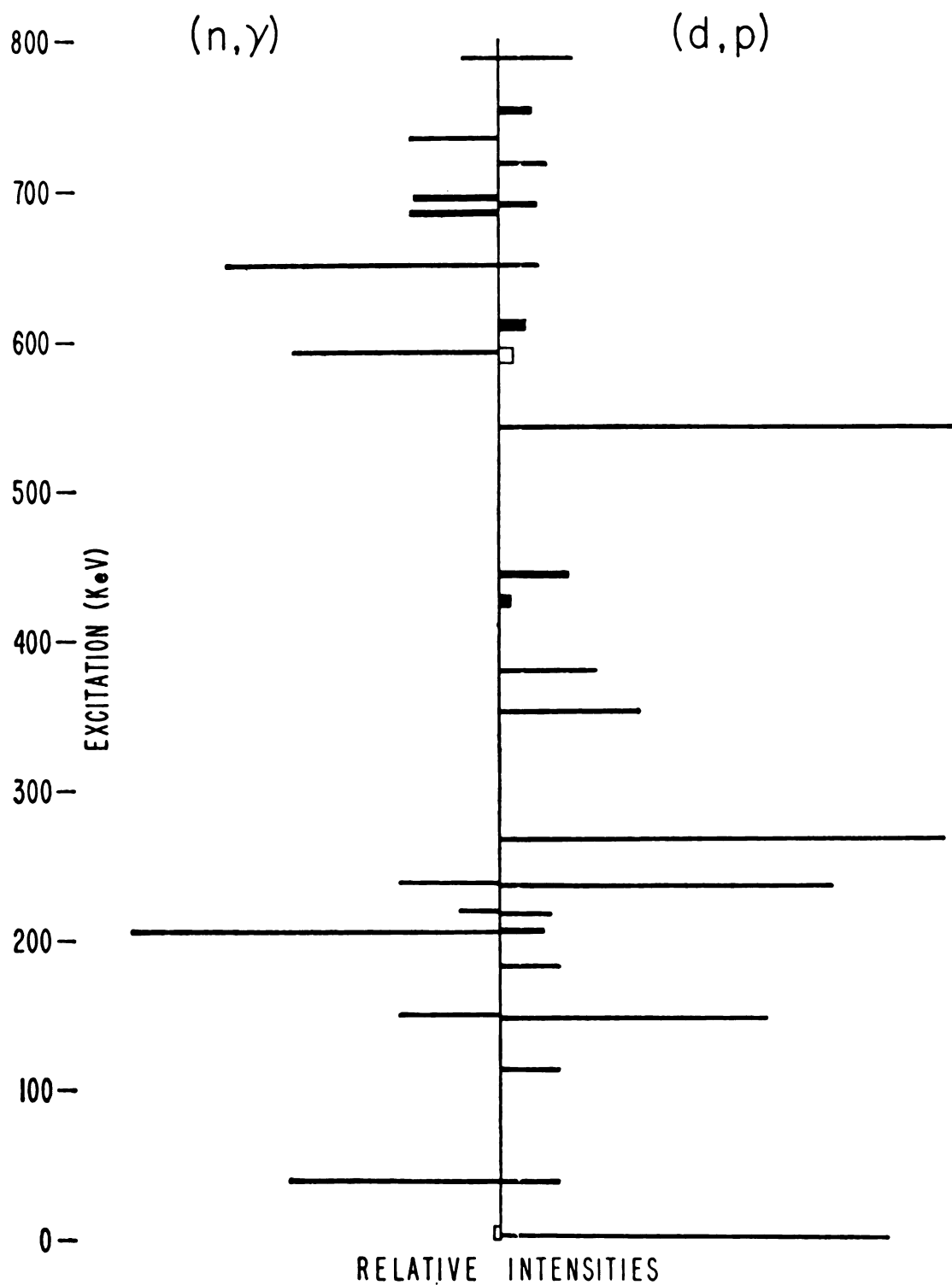


Figure 7

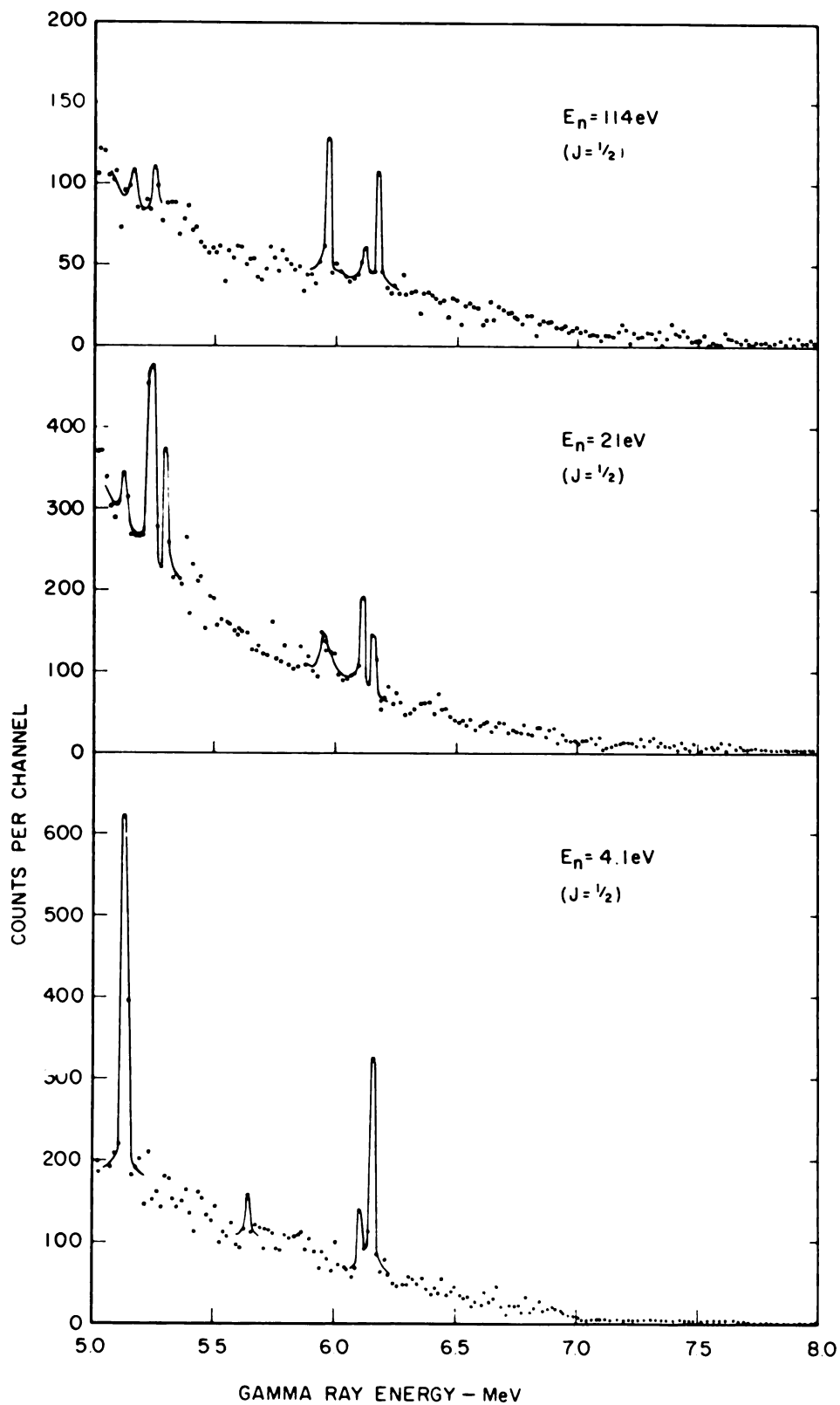
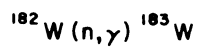


Figure 8

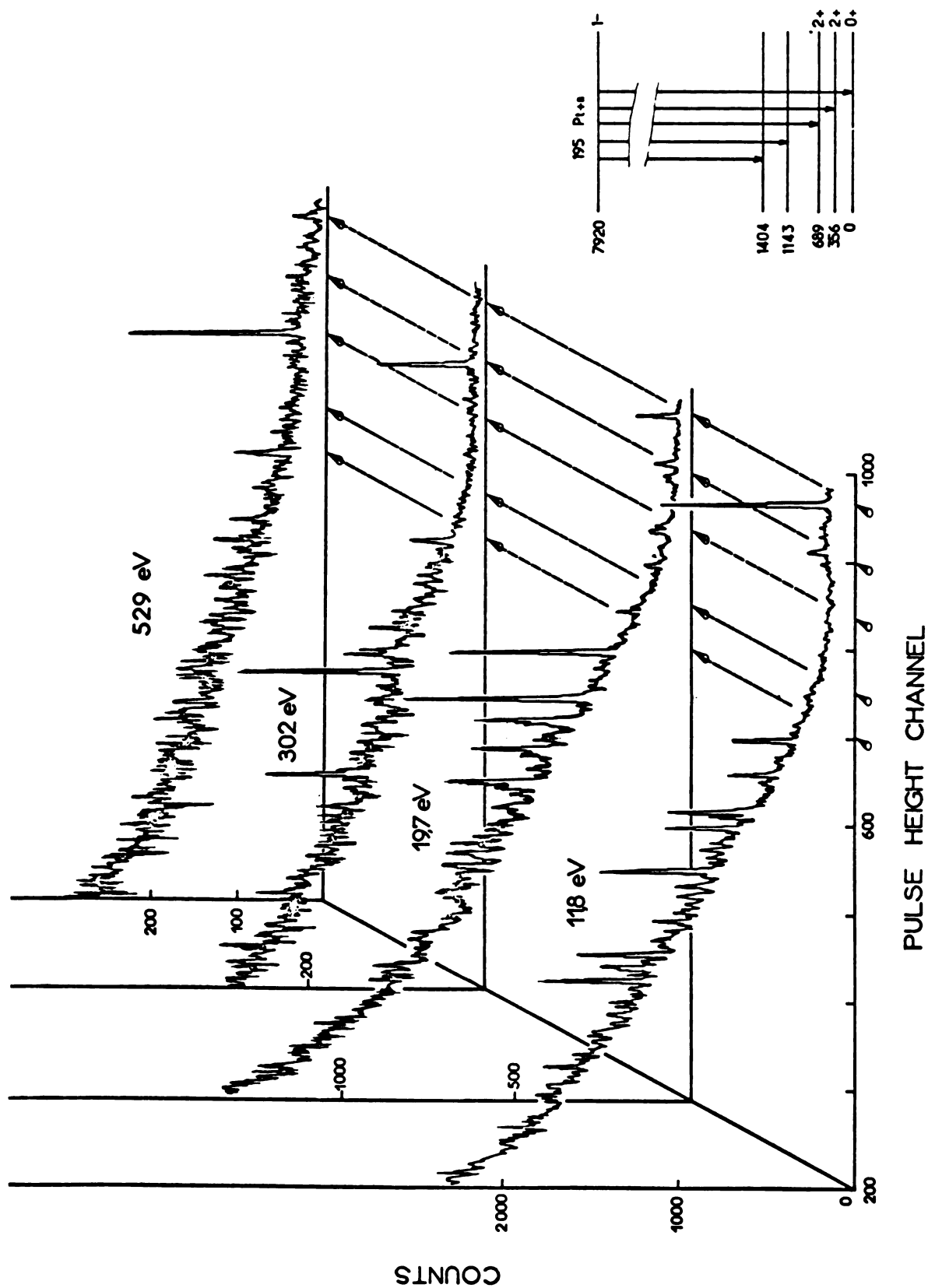
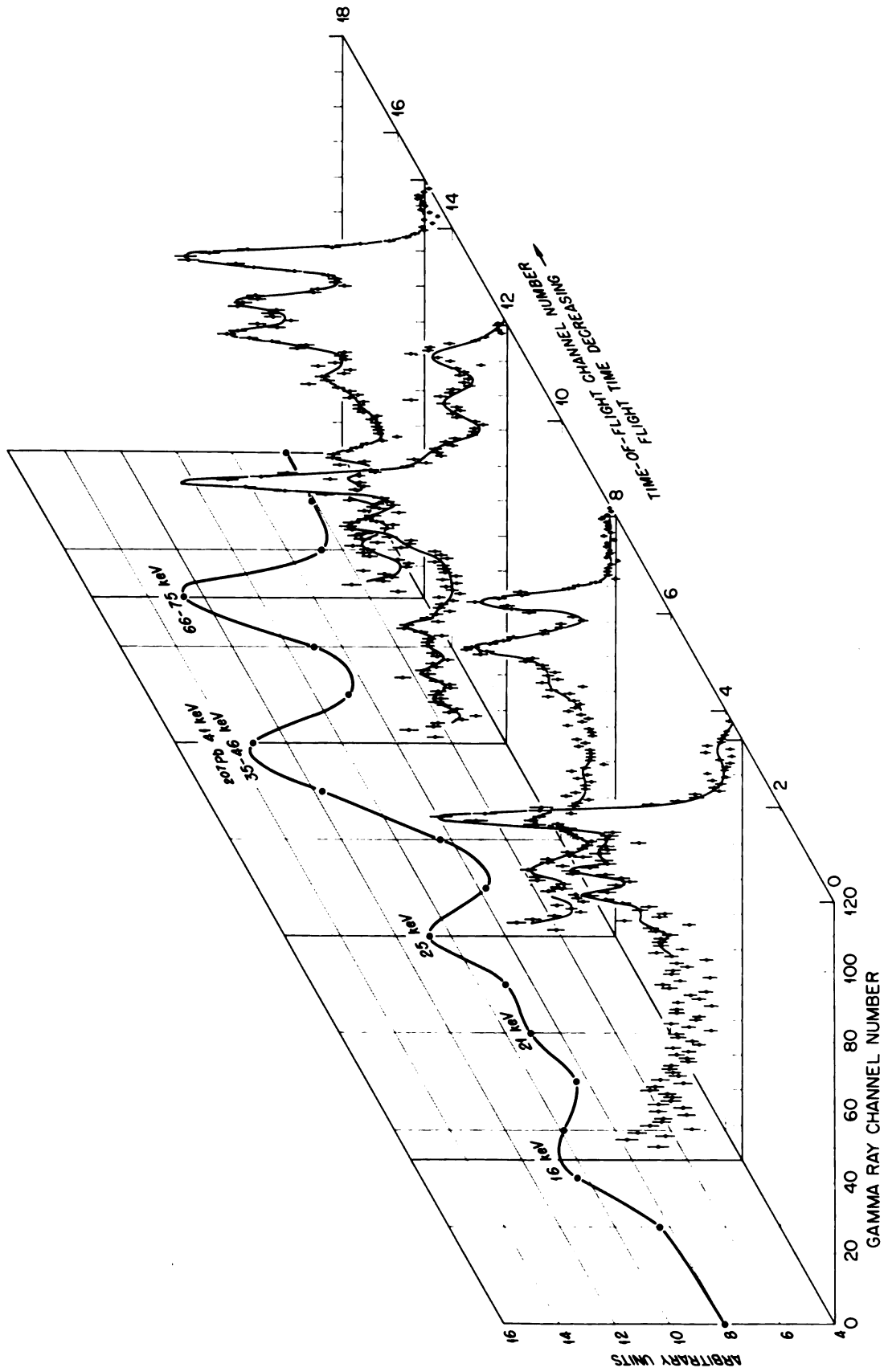


Figure 9



Yield of Gamma Rays as a Function of Neutron Energy for ^{206}Pb Target (Neutrons in 20 Channel, Gamma Rays in 128 Channel).

Figure 10

DISCUSSION
OF
PAPERS I.B. (Bollinger) and I.C. (Harvey)

Chairman: H. Maier-Leibnitz

Secretary: J. Manley

MAIER-LEIBNITZ: We have three written questions by Schult:

- 1) The multiplicity, the number of γ -quanta emitted per captured neutron, has been measured for many nuclei. It was found to be between 2 and 4. Is not this experimental result as well as the recent theoretical investigations strong evidence for the fact that the compound states are not "featureless"?
- 2) As stated, thermal neutron capture cross-section measurements are of value for nuclear physics when γ -ray spectra from these reactions are studied. The cross sections should then be accurate within about 6%. In the case of ^{162}Dy , the Russian value of 215 ± 16 b differs considerably from the U.S. value: 152 b with an error of less than 10%, as far as I remember. Does this example not show that we need better thermal neutron capture cross-section data, even if we do not study (n, γ) spectra at the same time?
- 3) What do γ -ray spectra measurements in many resonances (to allow for the Porter-Thomas variation in partial radiation widths, page 11 of Harvey's paper) tell you about "the specific character of the capturing state" (your Ref. 15)?

To start our discussion, I will ask Schult to choose one of these questions, the one which he thinks would contribute most or have the most general interest.

SCHULT: Well, my questions are directed to Harvey's paper. I am afraid they will not contribute so much to an answer of the question: "Why do we want to have strong neutron sources?" The study of neutron capture in resonances is, without any doubt, very interesting, but I think that we learn from this mainly something about the statistics of these processes. I question whether we can extract from these data information about the structure of the compound nucleus. I also wonder whether Mottelson meant, for instance, levels at 1.5 MeV when he was saying at the Gatlinburg conference that we understand the low-lying levels of nuclei. Should we make effort to understand levels at excitation energies around 6 MeV if we do not even understand the structure of states at 1.5 MeV?

MAIER-LEIBNITZ: Well, thank you. I think Motz is trying to prove his principle of written questions, so we have some questions. Will you give them?

MOTZ: To Harvey I would like to address the following comment. In the ^{170}Tm example,

the agreement that was referred to in the literature between the (d,p) and (n, γ) work was obviously with respect to energies of excited levels and not to intensities. Your remark that the Porter-Thomas distribution should apply to high energy gamma rays is entirely true when the low-lying final levels to which these distributions belong, even from a specific initial resonance, are not correlated. However, in the case of ^{170}Tm we believe from our assignments that the ground state and the first excited state are members of this same rotational band and, therefore, they have very common matrix elements in the nuclear wave functions. Now this is not entirely true because they are not pure Nilsson states; however, the intensities for, say, electric dipole transitions for these two states, would then involve essentially the same matrix elements and we can not understand why there would be such a large fluctuation in this case.

HARVEY: We went through similar arguments on the tungsten data since we also populated a rotational band just as you did in ^{170}Tm . I do not know the answer, but, as you know for the ^{182}W case, there was a large fluctuation from resonance to resonance. Hence, any explanation which you devise to explain the effect for thermal neutrons will have to be modified to explain the resonance data.

In reply to Schult's early comment, I would agree that for nuclei where the level scheme is complicated at low excitation then 1 MeV can be considered high excitation. Near the closed shells, where the level structure is quite simple, the "low" excitation can extend up to several MeV. For example, in nuclei near closed shells such as lead, the levels are well understood up to 3 or 4 MeV.

SCHULT: However, ^{182}W is not close to the closed shell.

HARVEY: Definitely, I agree. But I showed several slides on angular distributions from ^{208}Pb and capture in ^{208}Pb , and I think this is where we should devote more of the effort to start with to get things understood in a region where they are simple.

SCHULT: I completely agree with what you say regarding the capture in the resonances, but I think we are confusing two things which are different. In resonance capture studies you look for variations as in the Porter-Thomas distribution. What we are attempting is to understand the structure of the compound nucleus in more detail. The word compound nucleus is perhaps a little misleading. Most of us know that the multiplicity, the number of gamma quanta emitted per captured neutron, is usually between 2 and 4, mostly around 3. This seems to indicate that 2- or 3-quasi particle states will contribute only slightly to the wave function of the compound system of an odd-A nucleus. Thus, it would be interesting for us to understand, for instance, why one observes a strong primary gamma transition to the $I = 1/2$ level of the [521 $\frac{1}{2}^+$] band in ^{165}Dy and why the corresponding transition is missing in ^{163}Dy . In terms of the pairing model, one would expect that both target nuclei are similar. So what we have to consider is the detailed wave function of the compound system and I do not see how you can get an answer to this when studying the neutron capture in different resonances.

HARVEY: Well, I think anything you can say for thermal capture, I can say for resonance capture, and I can get more data by looking at individual resonances.

SCHULT: About the detailed information about the wave function? Can you write down the wave function!

HARVEY: No. I do not believe you are ever going to say that you have a perfect compound nucleus. The high energy gamma rays may be more indication of the doorway state or a simple configuration emitting a gamma ray. The bulk of the gamma rays that come off (the lower energy ones) probably arise from the compound nature of the capture state. I do not think neutron capture will be exclusively compound capture or strictly doorway-state capture.

CEULEMANS: You mentioned the tin experiments where you thought you were seeing some things in the doorway states. Now, I would say the primary transition would be the thing to look at, because there you might find something that looks like the prediction where you get high values of ν . If you look at the high energy transition, the transitions to the lower lying states, then you get in the compound nucleus picture, and you get ν equal to one. Have you had a chance to look at these primary transitions in tin?

HARVEY: I would say to the first part of your comment that I disagree with you. I have the picture that the high energy transitions are more likely to have three-quasi-particle doorway-state character, in which case the strengths of these gamma rays are enhanced. For compound nuclear states, the single particle strength of Weisskopf is spread over the compound states. Its strength is then reduced by the ratio of level spacing between compound states divided by the spacing of the single-particle states. This gives small M1 transitions. But in this doorway-state picture the strength does not get distributed out over all the compound states, but stays preferentially in a few states, which will then be strong. Those are the ones, then, from which the high energy gamma rays would show the greatest departure from a Porter-Thomas distribution. But, I think we completely disagree, don't we?

CEULEMANS: As far as I understood from the recent Gatlinburg meeting, the intermediate state is quite high and does not have a very large energy difference from the capturing state. You also showed the Duke results. There you have a gross structure which splits up into a fine structure. Suppose you have a group of resonances in which there is fine structure, but they belong to a gross structure group. Now, in these resonances, if you look for the primary transitions, then you should see that the intensity of the primary gamma rays does not fluctuate as much because they belong to the same gross structure group --because they belong to the same doorway-state transition.

MAIER-LEIBNITZ: We are getting a little deep in a detailed problem. I had hoped we would get swamped with proposals for new experiments in neutron physics in this discussion. I was going to ask, "Does anybody have a magnetic bottle for neutrons where you can keep them long enough to do a good measurement of a half-life?"

SMITH: Both speakers have omitted a rather important form of intense neutron source. This is a discreet source in both time and energy, for example, the (d,n), the (t,n), the (^3He ,n) processes. Using a small high intensity machine and multi-angle detection system, you can do a wondrous job of structure studies. You can delve into such things as analog states and doorway states, and spin dependence of stripping reactions, etc. You even can look into new nuclei; you do not have to look only into transuranium nuclei. (^3He ,n) is a gold mine in this area. I feel now that what we really need is a factor of 10 in energy and 100 in intensity. These are not astronomic factors, and, in fact, as regards Harvey's "quality" factor, I would make a change in this context; I would remove his dollar sign in the denominator and I would replace it with a cent sign!

JULIEN: In this Seminar devoted to intense neutron sources, we must not forget that the ability to utilize such neutron intensities can also be of great importance. Bollinger mentioned in his paper the improvement due to germanium detectors. Our old neutron capture experiments on plutonium performed with the Saclay linac using NaI crystals required five to seven weeks of operation. Detector resolution limited the reliability of results. Now we carry out the same experiments with a 6-cm³ Ge(Li) detector and the run lasts three or four weeks for equal conditions of neutron energy resolution. This is a typical example for which it is not necessary to increase neutron fluxes to obtain reliable information. People using intense neutron sources should not neglect to improve other parts of the experiment.

DABBS: There have been several statements indicating that the situation as regards neutron sources is good; let me indicate an area of inadequacy. In considering determination of the K quantum number associated with the transition states of fissioning nuclei similar to experiments now being made with oriented nuclei at the ORR reactor at Oak Ridge, it is clear that even such a linear accelerator as the new ORELA being built at Oak Ridge is far from being adequate as a neutron source.

SESSION II

CONTINUOUS REACTORS

CAPABILITIES AND LIMITATIONS OF HIGH FLUX
REACTORS UNDER DESIGN OR CONSTRUCTION

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ABSTRACT: This report presents the elements received by the author concerning the "Brookhaven high flux beam reactor" (HFBR), the German-French high flux reactor (GFHFR), the United Kingdom high flux beam reactor (UKHFBR), the Oak Ridge high flux irradiation reactor (HFIR), the Savannah River high flux demonstration (SRHFD), as well as some elements concerning SM-2, and the Argonne Advanced Research Reactor (AARR).

In order to study the possibilities of these reactors, a brief description of each is given, together with a list of their irradiation beam tubes, their actinide irradiation characteristics, the other equipment for irradiation, the available fluxes, the background noises in the beams, the space available in the buildings, the duration of the cycle, the annual expenditures of these reactors, and the present state of these installations.

Certain characteristics and limitations of reactors with emitted beams, of HFIR and of SRHFD, are examined, in the case of basic technology, just slightly modified.

1. POSSIBILITIES

1.1 BRIEF DESCRIPTION OF THE REACTORS

1.1.1 HFBR, Reference (1)

The basic concept of HFBR is to separate as much as possible the regions of maximum fast and thermal neutron flux. The external beam experiments require a large flux of low-energy neutrons; the extremity placed inside the reactor should thus be in the region of high thermal neutron flux. Many beam experiments, however, are sensitive to the background created by fast neutrons, so that the inner end of these tubes should avoid the region of high fast neutron flux. Separation of the fast and thermal neutrons is obtained by undermoderating the reactor core. A considerable portion (up to 60 per cent) of

the fission neutrons leaks into the reflector before being thermalized in the core. If the reflector material had good moderating properties and a low capture cross section, the slowing down of neutrons leaking from the core leads to a maximum thermal neutron flux in the reflector. The fast neutron flux is, of course, at a maximum in the region of the fuel. (Fig. 1)

The undermoderated core is made up of fuel elements uranium-aluminum plates of high uranium content (similar to the MTR-ETR line) which are heavy water cooled, this partly leading to undermoderation of the core. The reflector also consists of heavy water which, because of its low absorption, ensures a high ratio between the thermal flux and the fast or the epithermal flux over a large volume; the design of the reactor is thus simplified and problems due to heating, radiation damage, etc. that would arise with beryllium or graphite, are avoided. The heavy water that has just come out of the core is used for the reflector. The whole assembly is immersed in a vessel having a diameter of about 2.50 m in which the beam tubes are submerged.

1.1.2 GFHFR, Reference (2), (Fig. 2 and Fig. 2bis)

The GFHFR reactor has the same basic design as the HFBR, in particular in connection with use of an undermoderated core allowing an important part of its fast neutrons to leak into a heavy water reflector where end the beam tubes inside the reactor. In this case, however, the heavy water of the reflector is completely separated from the pressurized coolant (from 12 to 20 bars, according to whether the coolant is heavy or light water and according to criteria). This makes it possible to keep the heavy water of the reflector at a pressure just sufficient to avoid boiling in the hottest parts of the reflector; this pressure is low and is about the atmospheric pressure. Its main advantage consists in the increased possibilities of new types of experiments. However, this arrangement with separate fluids requires one or more partitions between the core and the reflector which have to withstand the coolant's pressure. This partition leads to an important loss both of reactivity (1.25 per cent for zircalloy) and a slight loss of useful flux (0.2 %). Most important, however, this wall is subjected to a very strong neutron irradiation (1.2×10^{22} fast neutrons/cm².year). The behaviour of the possible materials (zircalloy and aluminum alloys) to these irradiations is not well known and it is thus necessary to be able to remove the walls and to replace them.

The control rods are placed in the center of the core, thus making it possible to minimise flux deformation and to make the most

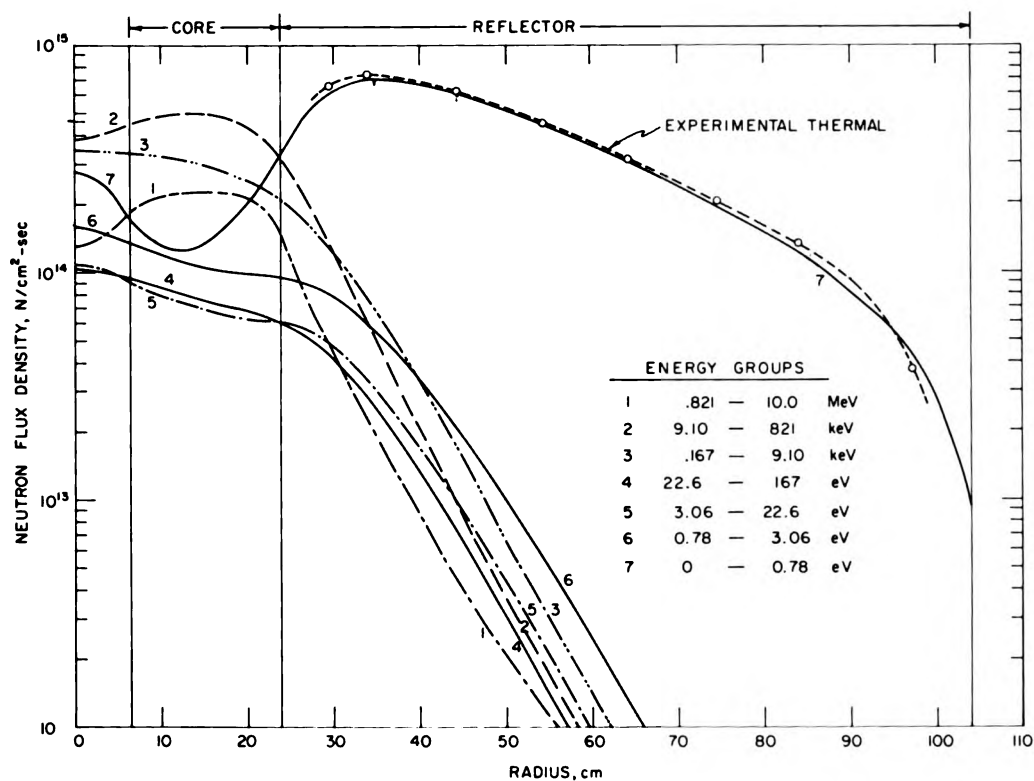


Figure 1 - H.F.B.R. Calculated and measured flux traverses.

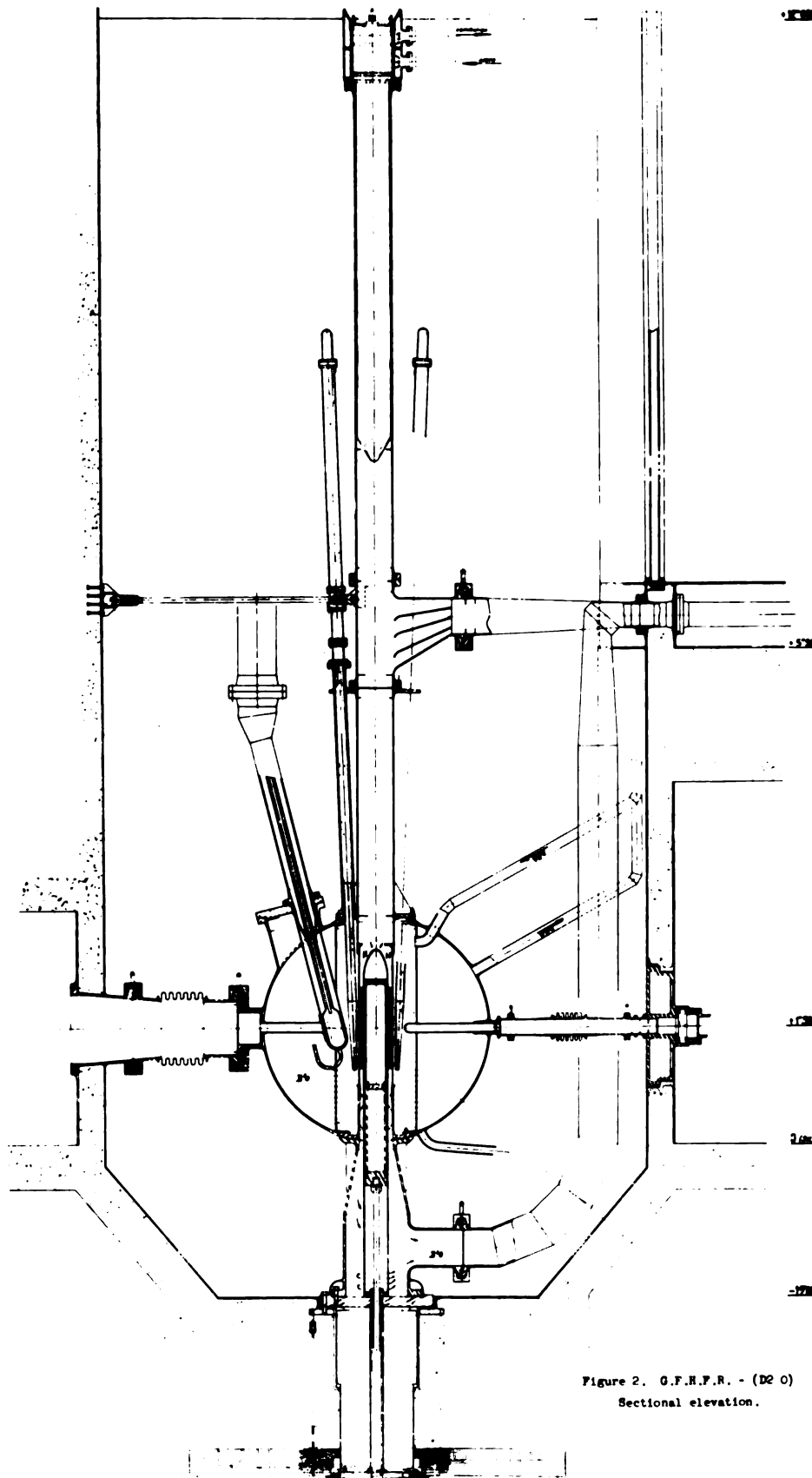


Figure 2. G.F.H.P.R. - (D2 0)
Sectional elevation.

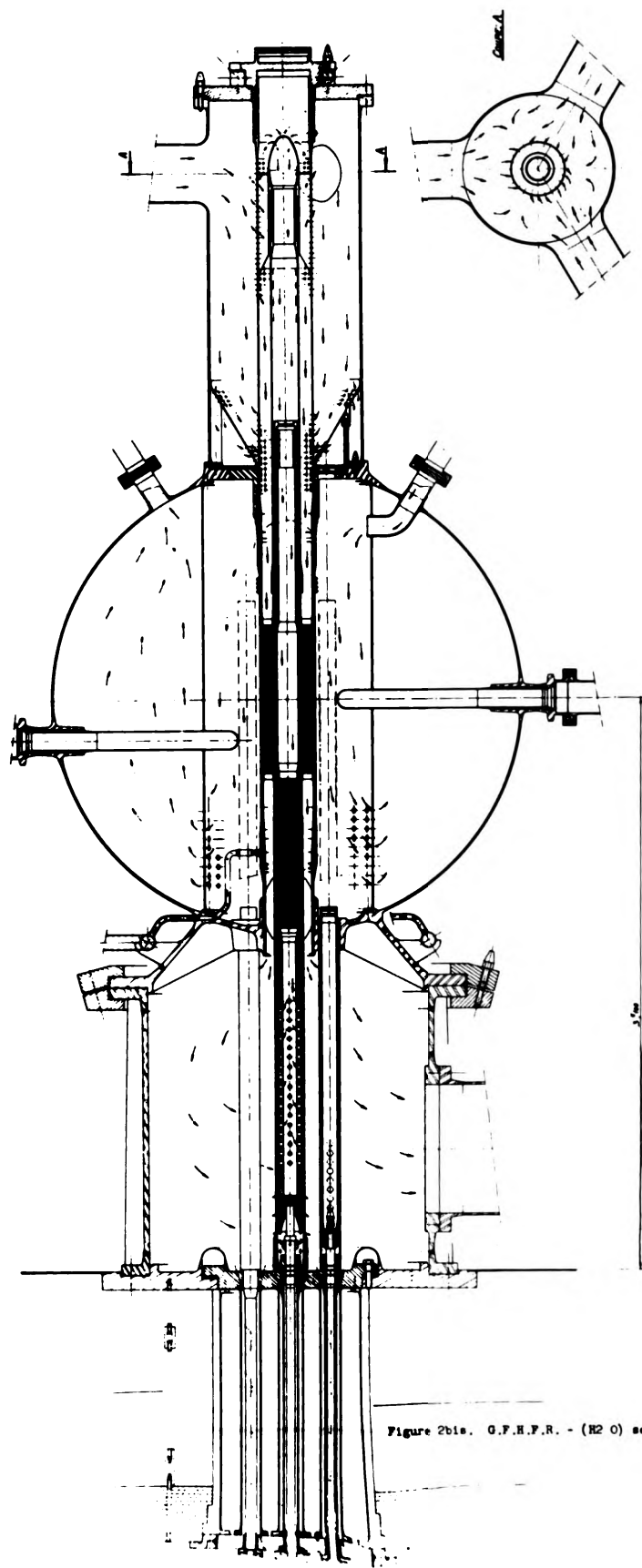


Figure 2bis. G.F.H.F.R. - (H2 O) sectional elevation

of the annular core. A flux of 1.5×10^{15} neutrons/cm².sec. is obtained at a power of 54 MW for a heavy water coolant, at a power of 56 MW for a light water coolant. The heavy water coolant is likely to be improved and it should be possible in this case, for instance, to obtain 2×10^{15} neutrons/cm².sec. at about 73 MW.

Furthermore, in order to facilitate the installation of original experimental facilities, as well as the transformation of existing facilities, the various experimental equipment of the reactors e.g. beam tubes, cold sources, hot sources, etc. are so designed as to be easily replaced, in particular using a light water pool in which the core and the reflector of the reactor are immersed. This principle of flexibility is also applied to the shielding in which practically the entire beam tube level at the pool wall is easy to attain and to modify. This makes it possible to place a part of the new experimental apparatus in the shielding recesses, to have the neutron conductors passing through the shielding and to be able to modify at a later date the spot where the beam tube crosses the pool's wall.

The reactor has an eccentric position in the building.

1.1.3 UKHFBR. Reference (3)

The UKHFBR is of the same basic design as the HFBR. The core and its coolant, formed by an assembly of tubes under pressure, are at about 34 bars, and are separated from the reflector which is nearly at the atmospheric pressure. This creates the same kind of problems as with the preceding reactor, so that the comments made before have to be applied here too. The control and safety rods are located immediately outside the core, at its periphery and consist of eighteen rods. At 100 MW, the maximum thermal flux obtained in the reflector is around 2×10^{15} neutrons/cm².sec.

The core, of a general cylindrical form, with a diameter of 46 cm, 75 cm high (with a maximum density of power of 1.8 MW/l), is formed of 37 fuel elements, each made up of ten concentric aluminum-uranium plates. The elements are placed in an aluminum block, which acts as multiple pressure tubes cut in the aluminum block. This latter should be able to be changed by retiring the upper thermal shield. (Fig. 3)

1.1.4 HFIR. Reference (4)

The concept of this reactor is that of a flux-trap in the center of a core. Three important factors determine the importance of the thermal flux in the flux-trap:

1. The quantity of neutrons leaking from the fuel into the flux-trap;
2. The slowing down and the absorption in the flux-trap;
3. The diameter of the trap.

In order to moderate completely, a big trap is necessary, but to obtain a high density of neutrons, a small volume is required; an optimum compromise thus exists. The best results for optimization of (2) and (3) are obtained with light water. The smaller the core, the bigger the leaking of neutrons of the core towards the trap. The limitation on the reduction of the core volume depends on the heat which can be evacuated and which cannot be greater than a few MW/l.

The HFIR reactor thus consists of three concentric and cylindrical regions: the central region where the flux-trap is located contains light water and has a diameter of 12.7 cm. The fuel element region located immediately around the flux-trap is formed of two cylindrical fuel elements (outside diameter: 43.4 cm), made out of curved vertical aluminum-uranium plates. The core is surrounded by a cylindrical beryllium reflector with a thickness of 30.5 cm. The control rods, having the form of concentric cylinders containing the poisons, are located between the exterior fuel element and the beryllium. The whole is placed in a tank with diameter of 2.40 m, this including a light water reflector, after the beryllium. The tank is submerged in a light water pool of about 4.80 m of diameter. (Fig. 4)

The actinide target to be irradiated is loaded in the core through the tank's lid, through the pool, which makes all handlings easy.

The beam tubes are sealed to and supported by the reactor pressure vessel, by means of a system of clamped and bolted flanged joints. From the flanged connection at the pressure vessel, each tube continues through the reactor pool and pool wall, and terminates in a recess located in a large beam port cavity in the shield wall. These devices facilitate the changing of these beam tubes.

1.1.5 SRHFD, Reference (6)

One of the production reactors of Savannah River has been adapted for operation at a high flux. In order to attain a high flux, it is necessary, in this case, for the absorption of the core to be very low and the specific power very high. The first condition has made the reactor operate at a very low fuel concentration: 8.5 kg of U^{235} in the 7.4 m^3 of the core (diameter 2.2 m, height: 1.8 m) i.e. 1.1 g/l which means about 1/50 to 1/100 of the same quantities as for conventional high flux reactors using aluminum technology. This is possible because this big reactor is moderated and cooled with heavy water, being formed of aluminum-uranium fuel elements, immersed into a heavy water tank of 3.6 m diameter and a height of 4.6 m (Fig. 5).

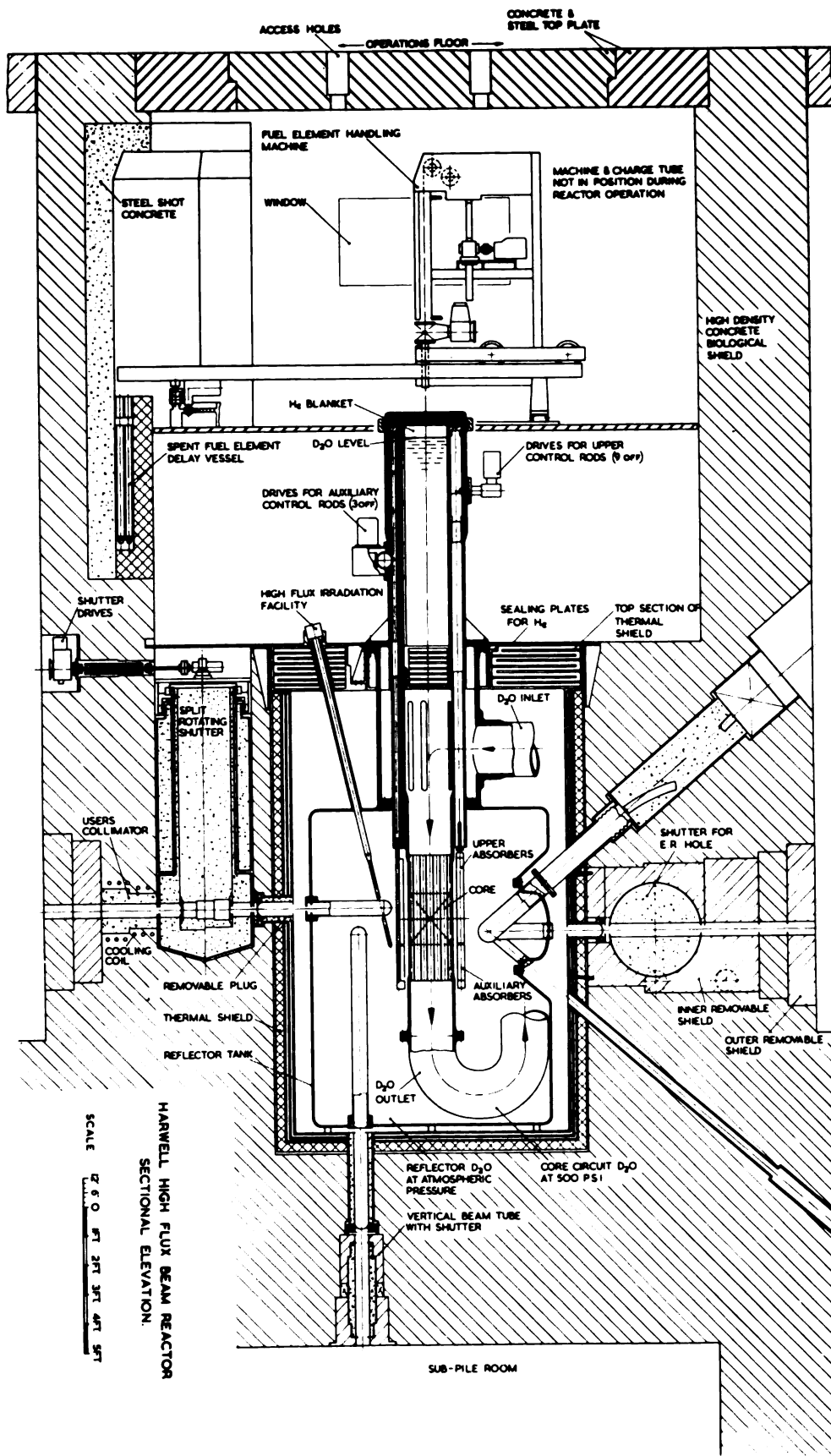


Figure 3

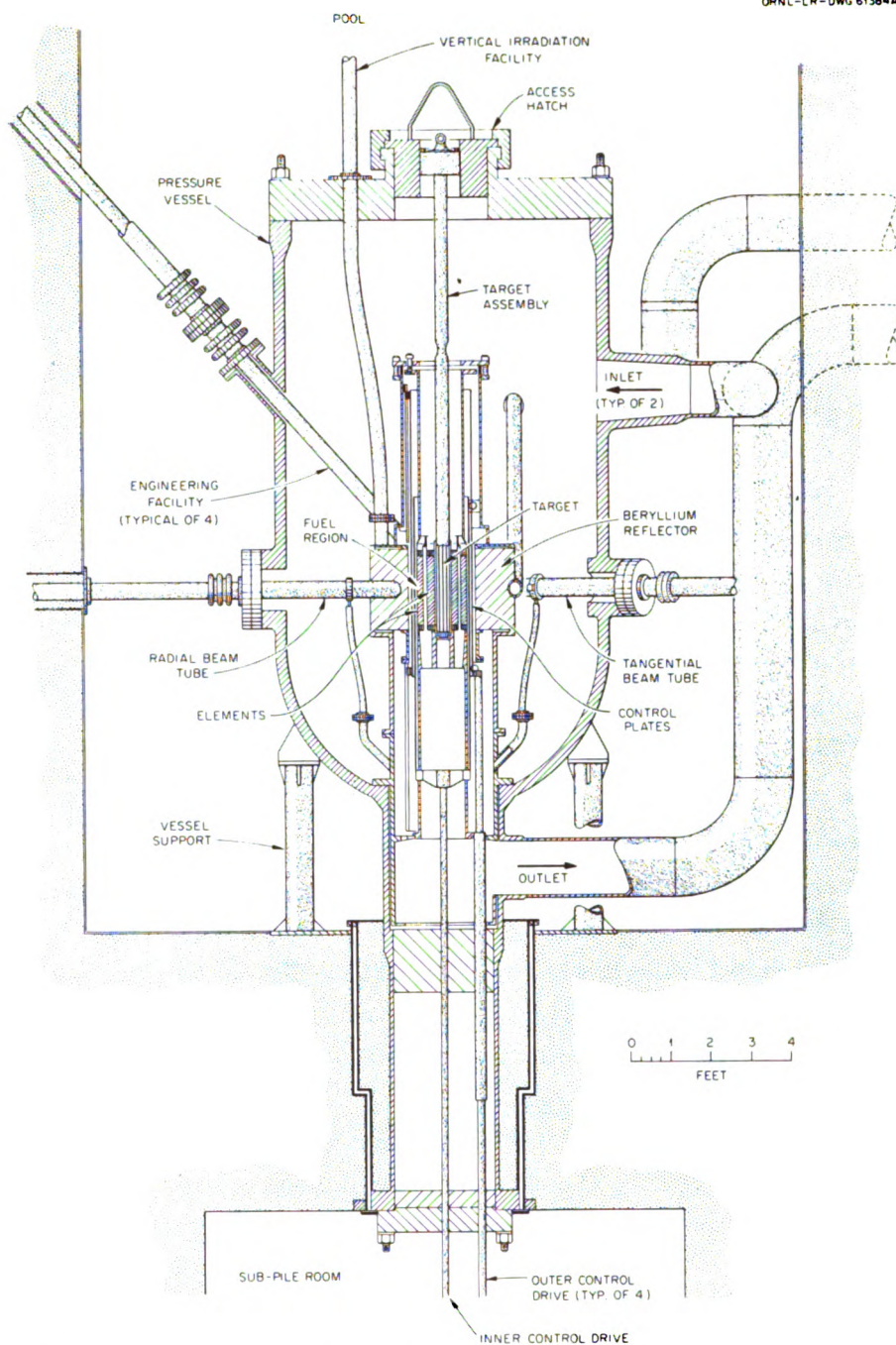


Figure 4 -;H.F.I.R. - Vertical Section of Reactor Vessel and Core.

The relatively high specific power for such a type of production reactor (of about 0.1 MW/l, i.e. around 1/20 of the medium specific power of high flux reactors) was obtained thanks to the high pumping capacities, the reduction of the diameter of the core (2.2 m) which has led to an increase in the speed of heavy water in the fuel elements up to 21 m/sec. and the relatively high heat transfer surfaces of the tubular elements employed. A further advantage of high speeds is that the temperature at which the local boiling begins is about 50° higher than the saturation temperature; this would make it possible to obtain fluxes of about 700 w/cm². Since the thermal neutron flux, averaged over the whole reactor volume, is around 2 to 3 x 10¹⁴ n/cm² for the conventional high flux reactors, the thermal neutrons averaged over the whole reactor volume attain in this case about 10¹⁵ n/cm².sec. The large volume of the reactor and the high specific power lead to a power of 735 MW. During the cycle that lasts for about a week, 40 per cent of the initial load of fuel is burned. These cycles are possible because the control covers a large reactivity margin and is able to distribute the neutron fluxes according to an optimal outline. The high flux region of thermal neutrons thus covers practically the entire reactor and the flux exceeds 10¹⁵ neutrons/cm².sec. in a volume of 6.7 m³.

The immediate applications of the Savannah River high flux charge are to assist the Transplutonium program, and to provide an irradiation facility for research programs being sponsored by several AEC research laboratories. Schedules are such that Savannah River can advance the program by about a year by pre-irradiating, the Pu²⁴² intended as target material for the HFIR. The high flux operations lasted from February to December 1965, by which time the target positions in the lattice had reached exposures of 40 to 50 neutrons/kilobarns.

1.1.6 The SM-2. Reference (7)

The reactor SM-2 is designed for carrying out various investigations in the field of nuclear physics, solid-state physics, physical metallurgy, radiation chemistry, reactor physics and engineering, and many others.

The reactor is a parallelepiped measuring 42 x 42 x 35 cm³, with an inner cavity of 14 x 14 x 35 cm³ which serves as a neutron trap (Fig. 6). The corners of the parallelepiped are occupied by the shim rods. The core is formed by 7 x 7 x 35 cm³ fuel assemblies and is surrounded by a reflector made from beryllium blocks interspaced by water. The whole is located in a vessel filled with light water of 1.5 m diameter, and 7 m high, pressurised at 50 bars. To carry out

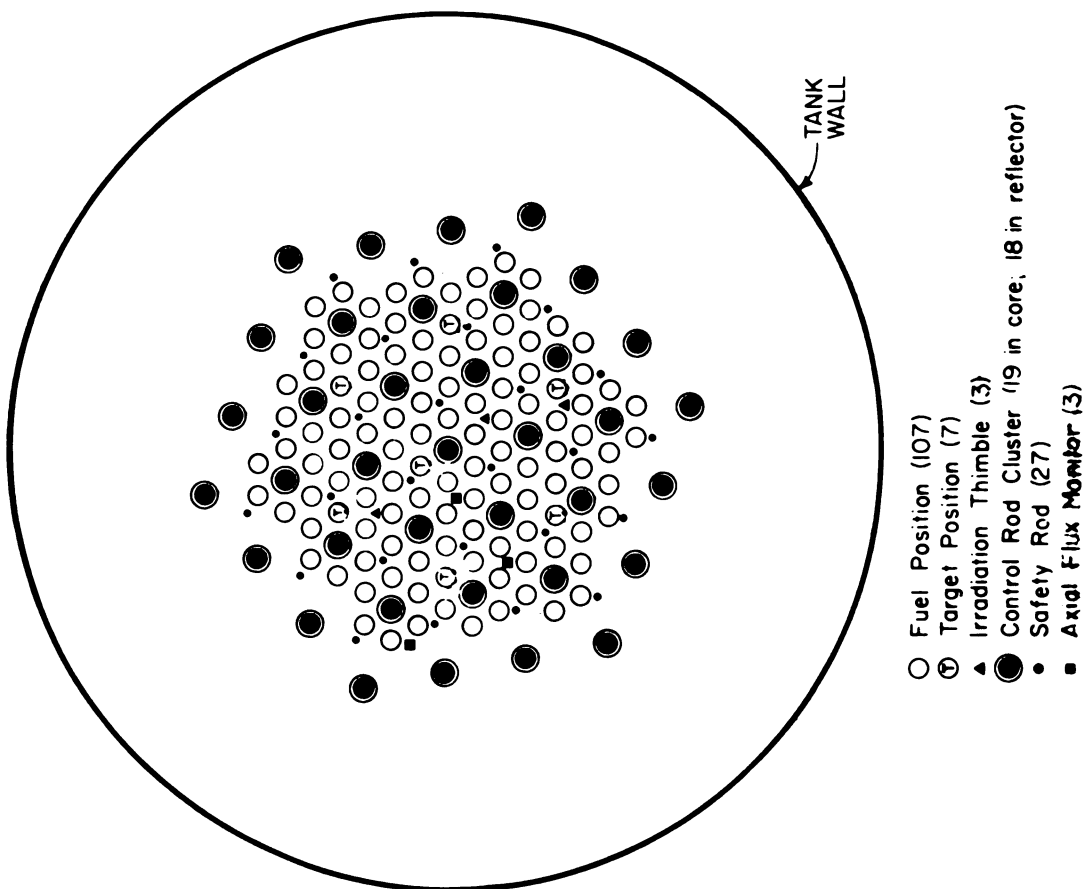


Figure 5 - SAVANNAH RIVER HIGH FLUX DEMONSTRATION

face map of high flux lattice

experimental work the reactor is provided with five horizontal beam tubes, one slanting and eighteen vertical channels. The fuel elements consist of stainless steel boxes containing each twice 27 plates of nickel and uranium alloy, with nickel clad. The power is of about 70 MW.

1.1.7 AARR. Reference (9). (Fig. 6bis)

In order to obtain long life for the fuel elements charge in the core of the reactor (which could attain 90 days) and in order to reduce the importance of the Xenon poisoning, thus making it possible to start up the reactor again after shut down during the greater part of its life, the quantity of fuel is very large: 60 kg of U^{235} for a volume of 78 liters. The energy region where the average neutrons are absorbed is of ten ev to thirty ev at beginning of the reactor's life. The fuel elements are thin plates (1 mm) containing a cermet of highly enriched UO_2 and stainless steel, clad with stainless steel, the coolant channel being equally thin (1 mm). The annular core is made up of forty-five rhomboidal fuel assemblies. It surrounds an internal flux-trap zone, or ITC, which will be used primarily for carrying out research in the field of nuclear chemistry and especially for production of transuranic isotopes. The core is surrounded by 30.5 cm of beryllium acting as reflector, where end a number of horizontal tangent beam tubes and through tubes, and some vertical holes for irradiation tubes. Between the beryllium and the pressure vessel, water is provided for shielding and for protection of the pressure vessel against gamma and neutron radiations. The entire reactor is immersed in a pool.

The reactivity of the reactor is controlled by a combination of a fixed neutron poison and a set of six shim blades which are in the interior of the reactor, and six blades for safety, one of which serves also for the power regulation. These last six rods are at the outer perimeter of the core.

The specific power is 5 MW/l at the most and 1.3 MW/l on the average. A zoning of the fuel is under study in order to reduce as much as possible the ratio of these specific powers.

1.2 THE BEAM TUBES

1.2.1 HFBR

The experimental facilities for the external neutron beams include nine horizontal tubes. The arrangements, positions and internal dimensions of these tubes in the high flux region are shown in Fig. 7. Of these nine tubes, five are made for thermal neutron beams. These

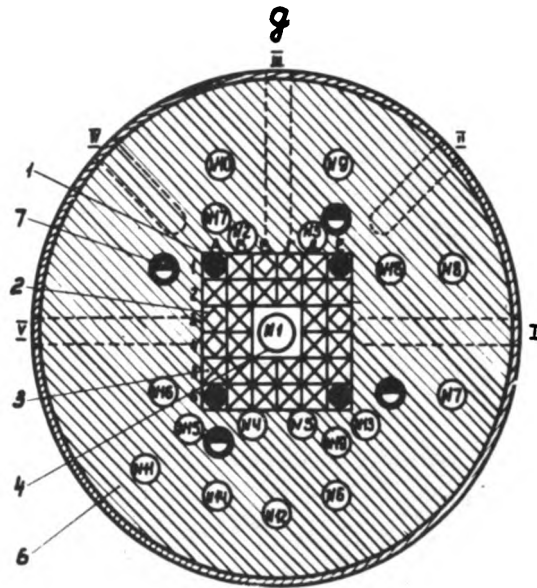


Figure 6 - SM-2 ; face map of the reactor

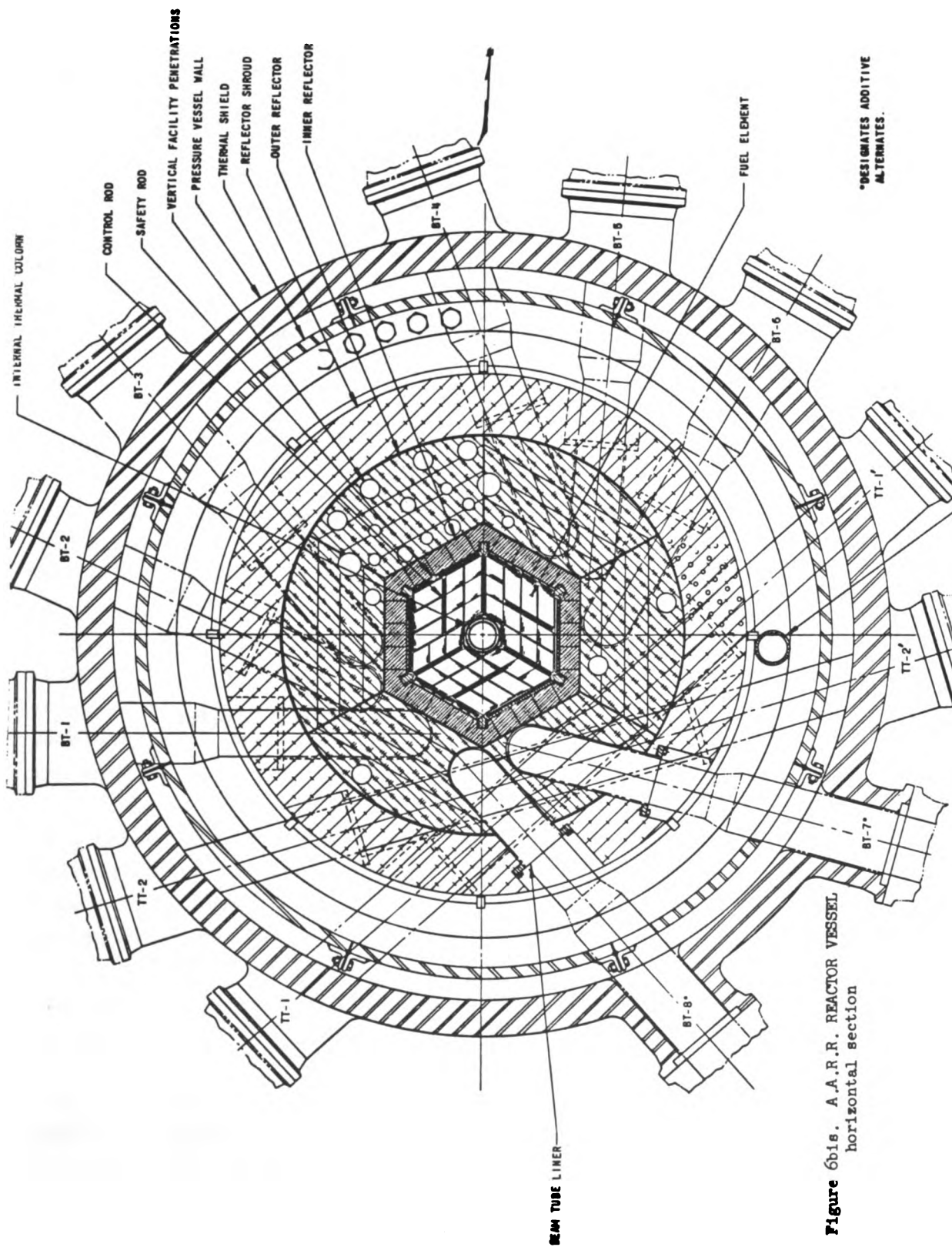
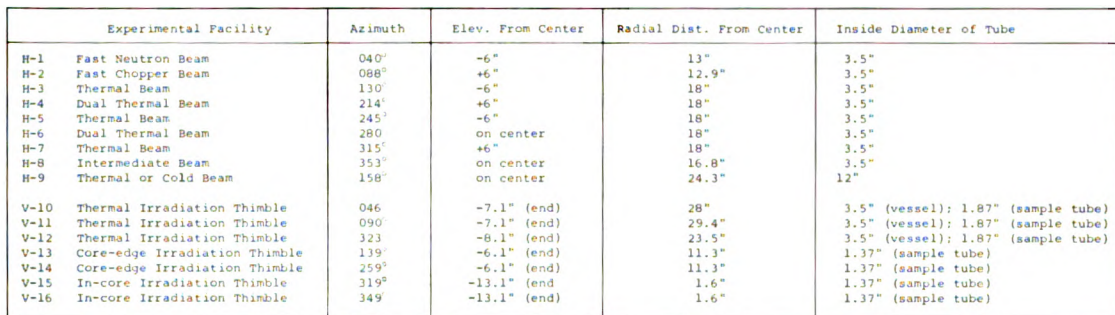


Figure 6bis. A.A.R. REACTOR VESSEL
horizontal section



- 133 -

tubes for thermal neutron beams are tangential, this being possible both because of the large distance from the flux maximum to the edge of the core, and because the maximum zone is fairly large, the maximum being flat. Two of these tubes have holes of a large diameter in the biological shield in order to be able to extract two neutron beams from a single tube, coming out of the reflector vessel. The H^1 and H^8 tubes (see Fig. 7) are tangential tubes located nearer the reactor's core than the thermal tubes, in order to increase the epithermal component in the neutron beams. The H^1 tube is very close to a corner of the core and, of the two tubes it is this one that will have the hardest thermal spectrum. The H^2 facility is a radial tube that reaches to within a few inches of the core. It will be used with a long flight-path fast-chopper apparatus. The H^9 facility is a tube with a large diameter which will be provided with a cold source.

1.2.2 GFHFR, (Fig. 8 and 9)

The number and the arrangement of the beam tubes will be reconsidered, once the work in progress completed. Let us note, for the time being, six tangential horizontal channels out of which three are viewing the hot source, a bundle of five neutrons conductors viewing the cold neutron source, one horizontal radial beam tube, one horizontal transversal beam tube, one sloping beam tube and one special beam tube for beams of fission products. There will be some additional small vertical or slightly inclined thimbles, entering the reflector vessel from the pool surface, for irradiation purposes.

Each beam tube, or cold source, hot source, etc. will be made easily replaceable. It is not only required from the experimental workers in order to attain flexibility, but it is also necessary due to high thermal stresses, to radiation damage in the tube tips, transmutation, etc. Handling of the vertical experimental facilities (cold neutron sources, hot neutron sources, etc.) will be done from above, using the pool.

We have noticed before that the shielding around each horizontal beam tube is removable, forming an experimental recess. This makes it easier to have access directly to the flanged joint of the beam tubes that have to be changed. The portion of the wall acting as separation between this recess and the pool can also be removed. Experimental equipment (collimators, etc.) could be set up in this recess.

As a result of much research work: hydrodynamic of liquid hydrogen vortices, theoretical neutronic in liquid hydrogen and liquid deuterium, experimental studies on low power piles, studies with pulsed

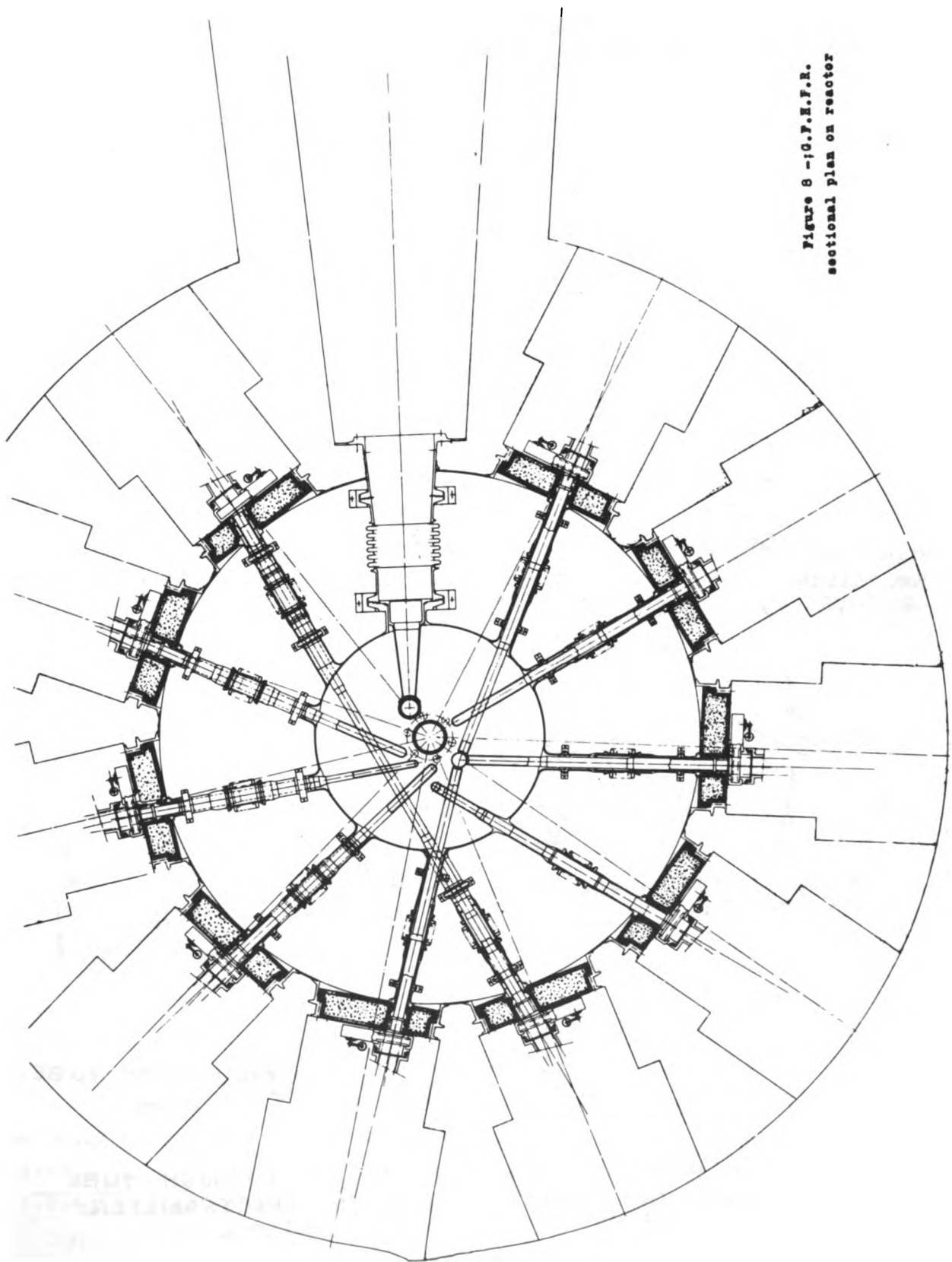
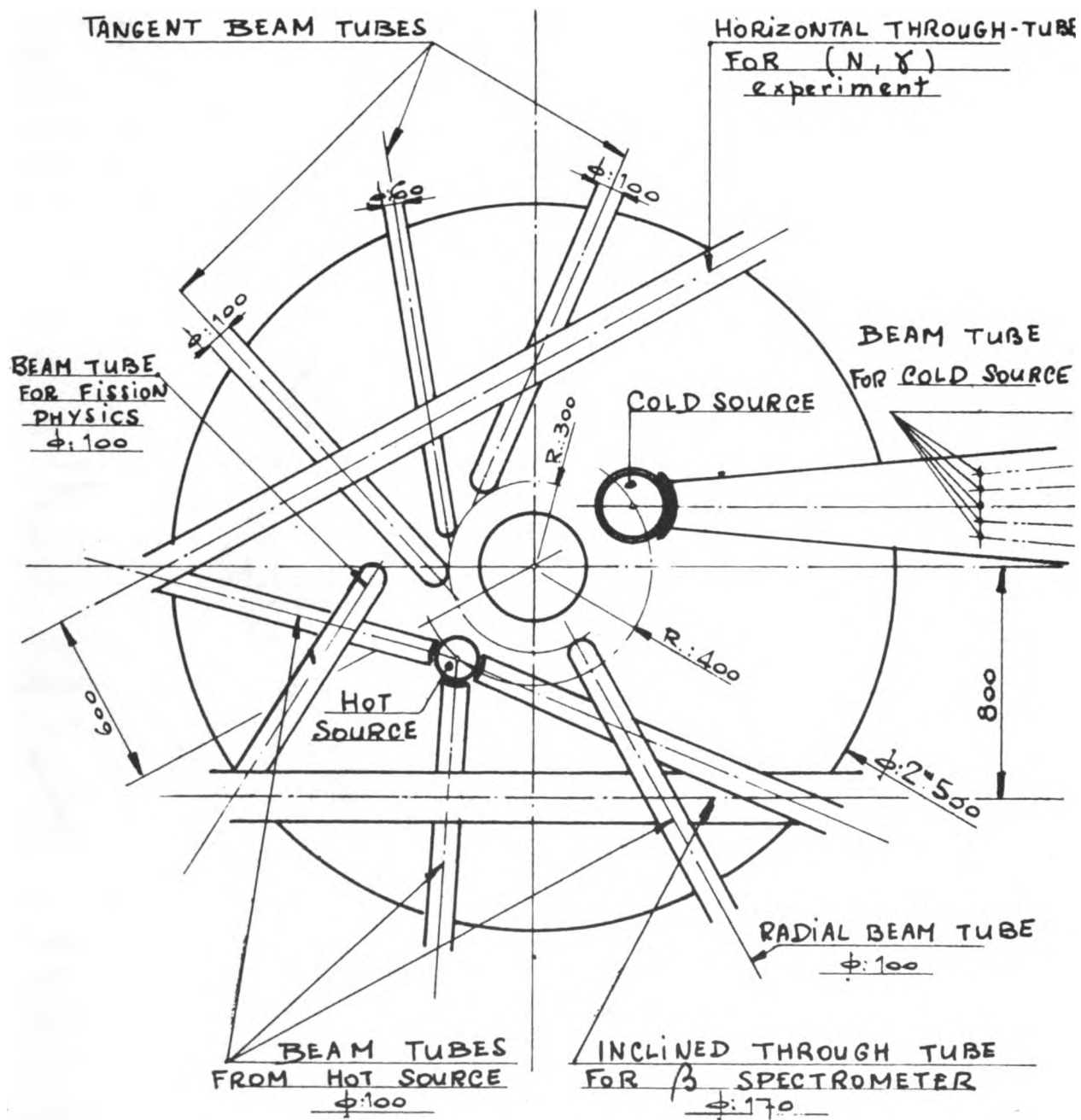


Figure 8 - J.O.P.N.F.R.
sectional plan on reactor

SOLUTION I et II BEAM TUBES IN THE REFLECTOR

Figure 9 -; G.F.H.F.R.
sectional plan on reactor tank
showing beam facilities



neutrons, etc., several types of cold sources have been envisaged, particularly a vortex-type source using liquid hydrogen and a conventional source using subcooled liquid hydrogen (or deuterium). Preliminary results lead to the conclusion that the greatest gain in cold neutrons would be obtained with a large diameter (40 cm) deuterium source, the center of which would be viewed by the neutron conductors. But the heating of such a source would lead to big technical and financial problems.

A study is also being made of a hot source composed of a graphite block heated to about 2000°. An important research program is now under way which includes a study on an installation in FR².

1.2.3 UKHFBR

There are five horizontal tangential beam tubes and one horizontal radial, all viewing the maximum flux of the reactor, i.e. 2×10^{15} neutrons/cm².sec. Four other horizontal beam tubes are connected with the cold and the hot sources (see further on). Farther from the core, but still in a region of very high flux, twelve tubes are located, sloping to the horizontal and vertical, as well as one through tube. All isolated tubes can have diameters up to 15 cm and can be removed from the reflector vessel, making it thus possible to adjust various diameters and lengths, in order to regulate as well the spectra as the collimation angles. The arrangement and the angles of the tubes cannot be varied, because of their being laid out with permanent holes in the thermal and biological shields (see Fig. 10 and 11).

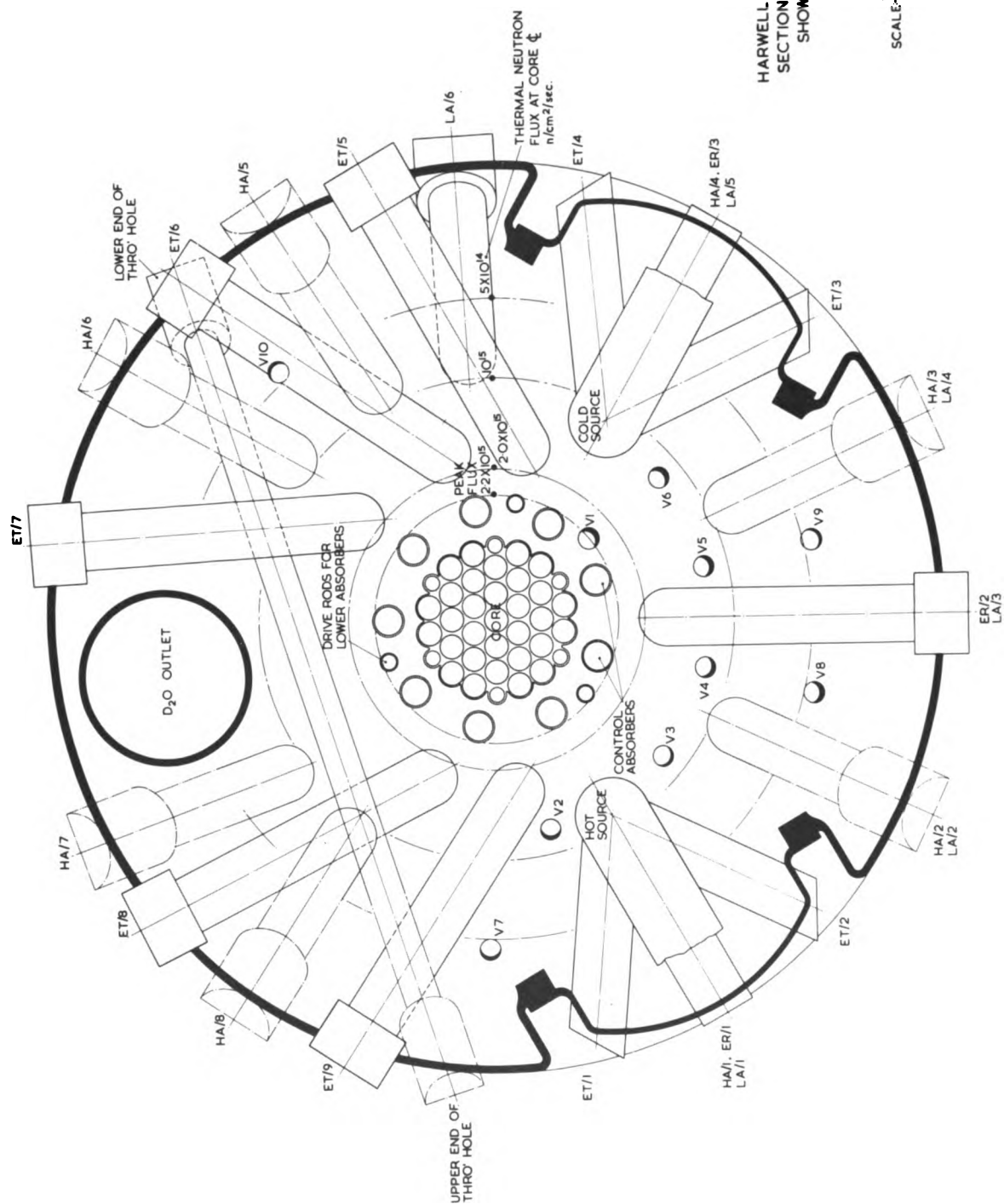
Fig. 11 shows all the tubes with their maximum 15 cm diameter. In the effective reactor, the diameters vary from 5 cm onwards. It may be that a small number of low flux tubes will be cut out, once a sufficient experiment will be obtained on the main experimental facilities.

The cold source of the UKHFBR, viewed by four tubes, would be loaded by a fifth having the biggest inclination. The set forms a multipode, connected with the reflector vessel. Liquid hydrogen has been envisaged for this cold source, and there are experiments taking place presently on the LIDO reactor, in order to get the necessary information for an optimum geometric design.

For the other multipode, graphite or beryllia blocks, maintained between 1000°C and 2000°C, are suggested. Experiments take also place on LIDO.

1.2.4 HFIR

Three horizontal beam tube experimental facilities are situated in the median plane, (Fig. 12). One of these beam tubes is



KEY -
 HA HIGH ANGLED
 LA LOW ANGLED
 ET EQUATORIAL TANGENTIAL
 ER EQUATORIAL RADIAL
 V VERTICAL

NOTE -
 BEAM HOLES ARE SHOWN AT
 THERE MAXIMUM DIA OF 6"

Figure 11
 HARWELL HIGH FLUX BEAM REACTOR
 SECTIONAL PLAN ON REACTOR TANK
 SHOWING BEAM & IRRADIATION
 FACILITIES

SCALE - 0' 3' 6' 9' 1FT 2FT

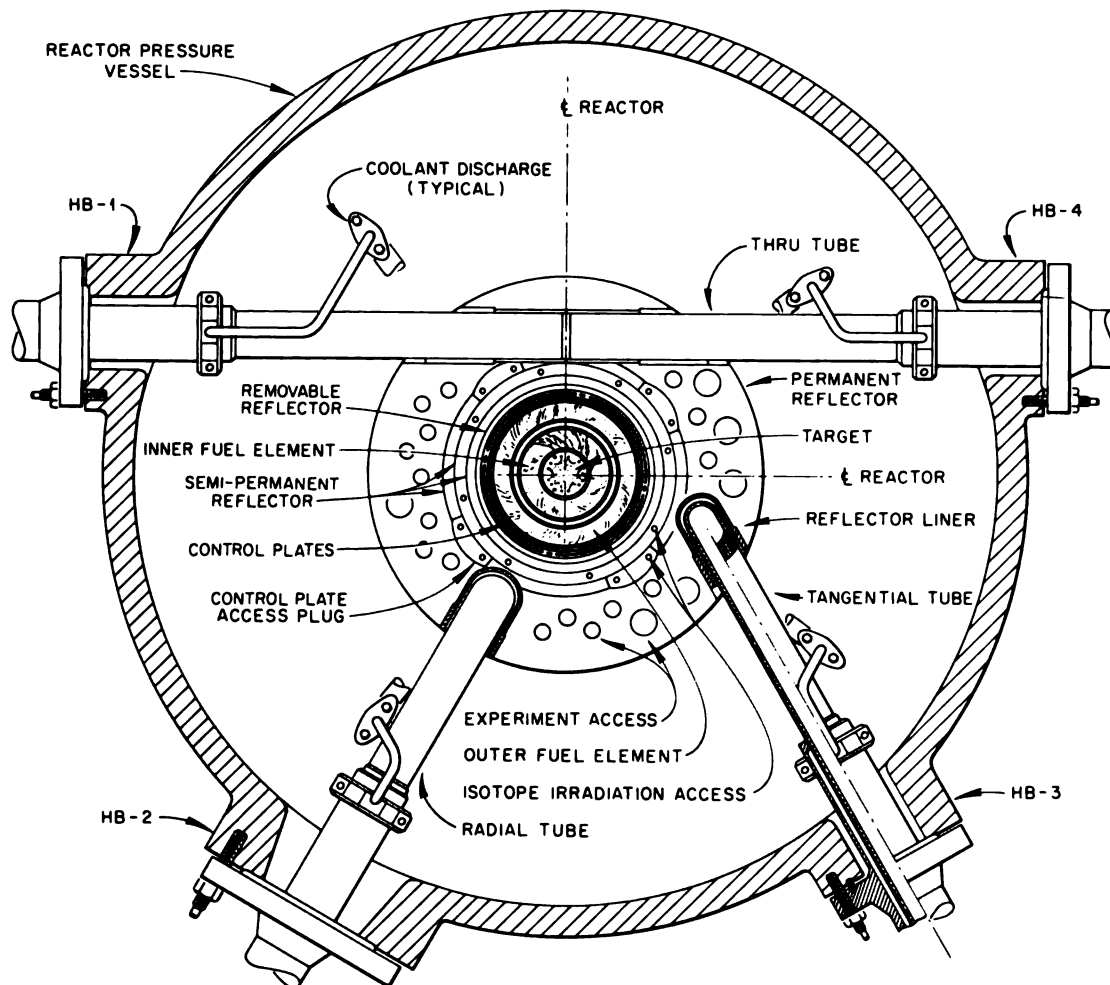


Figure 12 -;H.F.I.R. Plan View of Reactor Core.

radial, its extremity reaching into the permanent beryllium reflector. Another is tangential, passing 26 cm from the reactor's axis (the reactor's radius is about 21 cm). It penetrates also into the permanent reflector. The last tube is aligned along a tangential line, at about 38 cm from the core's axis, its two extremities stretching outside the reactor. It should make it possible to install two individual facilities or a single through tube.

1.2.5 SM-2

Five tubes are placed as shown in Fig. 6. They are made of zirconium and can be replaced.

1.2.6 AARR

The reactor has six horizontal beam tubes of a 12.7 cm diameter and two through tubes with the same diameter. It is intended that each "blind" tube will provide two separate beams of neutrons at the working face of the lateral biological shield.

Possibly, local neutron flux traps (H_2O) will be positioned around the tips of the "blind" beam tubes. The effect of such flux traps on reactivity and on the level of the neutrons flux will be measured in the AARR critical facility.

1.3 TRANSURANIUM ELEMENTS

1.3.1 HFIR

At HFIR, the target irradiations containing 300 grammes of Pu^{242} and smaller quantities of Am^{243} and of Cm^{244} would produce, annually, about 1 gramme of Cf^{252} with smaller quantities of berkelium einsteinium, fermium as well as other isotopes of californium.

1.3.2 SRHFD

At SRHFD, the use of a large useful volume would make it possible to produce 3 kg of Cm^{244} per irradiation of the Pu^{242} and Am^{243} target.

In particular, 520 g of Pu^{242} have been irradiated, in order to provide Cf^{252} in milligrammes for separation in the TRU Oak Ridge facility, reference (6).

An attempt could be made to define the capacity of such a type of reactor. When the target material is added, uranium has to be added also, in order to preserve the criticality; this leads to a lower flux. The average working flux, in terms of the cm^2 of absorbant, is given in Fig. 13. To indicate the importance of the effect, it can be noted that 1 kg of U^{235} represents 1400 cm^2 of absorbant, and 1 kg of Pu^{242} represents 50 cm^2 . With the present load of 0.5 kg of Pu^{242} , it is possible to operate at nearly maximum flux. It would be

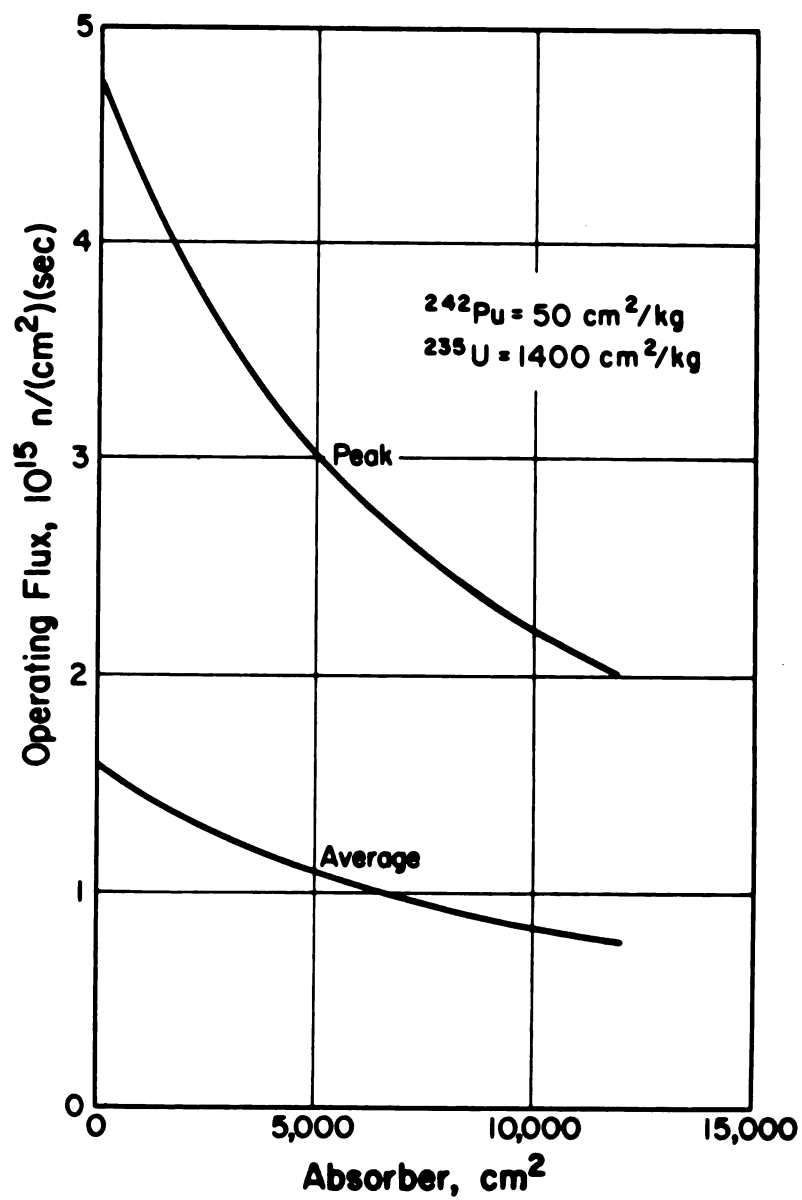


Figure 13 - SAVANNAH RIVER HIGH FLUX DEMONSTRATION
effect of target on flux

necessary to use 130 kg of Pu^{242} in order to lower the average flux during operation to below 10^{15} neutrons/cm².sec.

1.3.3 AARR

The internal thermal column (ITC) includes light water for the moderation and for cooling. Its radius is 7 cm and its height 45 cm. It will include five hydraulic rabbits of a 1.27 cm diameter, four static facilities of a 1.27 cm diameter and, later on, gas operated rabbits.

1.4 OTHER IRRADIATION DEVICES

1.4.1 HFBR

Seven vertical thimbles are fitted, of which two are in the center of the core, two in the reflector near the core and three in the reflector (Fig. 7).

1.4.2 GFHFBR

A study is being made of the projected devices.

1.4.3 UKHFBR

A small number of irradiation positions is being considered for the core and the reflector; their final number will depend on the future experiments and, particularly, on the reactivity absorbed. A typical load for studies of damage caused by fast neutrons would include four central holes (diameter 2.5 cm) of fuel elements, and one hole of a 5 cm diameter in the central fuel element. Inside the reflector, a hole of diameter 2.5 cm will be made on the peak of the flux, together with a number of holes or rabbits in the flux region of about 10^{15} neutrons/cm².sec. (Fig. 11). It is also planned to place rabbits in the flux zone of 5×10^{14} neutrons/cm².sec. where the ratio of thermal flux to epithermal flux is particularly large.

1.4.4 HFIR

It is planned to install in the future four engineering facilities not requiring a very high flux. Furthermore, the permanent reflector is perforated with 22 vertical holes that cross completely the beryllium. Sixteen similar holes are to be made in the replacable beryllium.

Besides the static irradiations, the vertical holes will make it possible to install hydraulic facilities. Finally, in the central target zone it is planned to set up a small hydraulic tube for the irradiation of small samples. This tube would be placed vertically on the central target zone and would allow short irradiations in the very high flux of the center without interfering with the operation of the reactor. The hydraulic system is already designed and is presently being tested out of the reactor. The installation will be made following

completion of the start-up program for the reactor. The estimated cost of this facility is about \$ 50,000.

1.4.5 SM-2

There is a vertical loop in the central cavity of the core, loops and positions for capsules in the reflector, in all nineteen positions.

1.4.6 AARR

Nineteen vertical irradiation facilities are placed in the reflector, i.e. seven of a 3.8 cm diameter (two 1.27 cm and two 2.54 cm hydraulic rabbit facilities are installed in four of the 3.8 cm vertical facilities), four of a 5.1 cm diameter, six of 6.3 cm and two of 7.6 cm diameter.

1.5 AVAILABLE FLUXES

1.5.1 HFBR

The maximum perturbed thermal flux in the reflector is 0.7×10^{15} neutrons/cm².sec. (see last measurements: 0.52×10^{15} neutrons/cm².sec.), reference (8).

1.5.2 GFHFBR

The maximum non perturbed calculated thermal flux is 1.5×10^{15} neutrons/cm².sec. The beam tubes and other experimental equipment produce both a general and a local decrease. A calculation based on the homogenisation of aluminum and the void of the beam tubes show a general decrease of 10 per cent in the flux. Measurements carried out in Alize around the tube's extremity show a local lowering of 18 per cent for a tube of 100 mm diameter. An attempt will be made to obtain a perturbed flux of 10^{15} neutrons/cm².sec. at the extremities of the simple beam tubes.

1.5.3 UKHFBR

The maximum non perturbed thermal flux that has been calculated, is 2×10^{15} neutrons/cm².sec. The three-dimensional Monte-Carlo program GEM has been used for the calculation of the beam tubes flux depressions. For a 10 cm diameter beam tube, the results yielded a flux in the beam tube of 0.9 of that for the same position in the reflector with no beam tube. For a multipode consisting of four beam tubes, the corresponding figure was 0.6.

1.5.4 HFIR

At HFIR, the characteristic fluxes are the following:

- Maximum unperturbed thermal flux in island 5.5×10^{15} neutrons/cm².sec.
- Average thermal flux in target
(300 g of Pu²⁴²) 2.0×10^{15} neutrons/cm².sec.

- Average non thermal flux in target 2.4×10^{15} neutrons/cm².sec.
- Maximum unperturbed thermal flux in
Be reflector: Beginning of fuel cycle 1.1×10^{15} neutrons/cm².sec.
End of fuel cycle 1.6×10^{15} neutrons/cm².sec.

1.5.5 SRHFD

The capacity of the control rod system (about 27 per cent) allows the initial uranium charge to be slightly less than twice the minimum required for criticality. Thus, with fuel operating at constant power, flux levels are roughly doubled from beginning to end of the cycle. The spatial distribution of flux in this big reactor (a diameter of around 2.20 m) is a chopped cosine, both axially and radially. There are further differences between flux in fuel positions and flux in target positions. But these differences are small (about ten per cent).

In view of the variations just mentionned, the term flux must be qualified in several ways. The flux of the center of the reactor is of 4.8×10^{15} neutrons/cm².sec. During a cycle, its average is 3.2×10^{15} neutrons/cm².sec. The mean flux in the reactor is 2.2×10^{15} neutrons/cm².sec. Its average, during the cycle, is of 1.6×10^{15} neutrons/cm².sec. Because of the short duration of the cycles, stoppage times are rather important. The former flux, averaged over both operation and shut down is 1×10^{15} neutrons/cm².sec.

1.5.6 SM-2

The flux in the central cavity is 2×10^{15} neutrons/cm².sec. The five beam tubes are arranged according to (Fig. 6). The three beam tubes (I, III and V) next to the core but separated from the latter by a 10 mm thick sheet of water have at their ends a flux of 0.5 to 1×10^{15} neutrons/cm².sec. The two beam tubes II and IV, separated from the core by about 70 mm of beryllium, have a variable flux, since they are opposite the control rods. The beam tubes I, III and V would have 2.2 times more flux with a sheet of water of 40 mm and 1.8 times more flux with 100 mm of Be.

1.5.7 AARR

The unperturbed flux in the ITC is computed to be approximately 3.7×10^{15} neutrons/cm².sec., the average in a typical ITC sample being 1.7×10^{15} neutrons/cm².sec. The unperturbed peak thermal neutron flux in the beryllium is estimated to be 1.10^{15} neutrons/cm².sec. The thermal neutron flux levels in the zone of the vertical irradiation tubes, in the beryllium, will be much in excess of 10^{14} neutrons/cm².sec.

1.6 BACKGROUND NOISE IN THE CHANNELS

1.6.1 HFBR. Reference (8)

The experiments carried out at Brookhaven have shown that if the neutron beam tubes were placed tangentially to the cylinders centered on the core's axis, the fast neutrons and gammas coming from the collimators of these tubes must have undergone collision with scattering at about 90° in the region of the tubes nose. This tangential arrangement reduced the fast neutron intensity by a factor of 30 compared to the case of a radial tube. The anticipated reduction for the gamma background was still larger: a factor of 100. The measurements made during the start-up of HFBR confirm this point of view. A few results are given in table I.

TABLE I

	: H ¹ channel : (epithermal)	: H ² channel : (fast)	: H ⁴ channel : (thermal)
Distance from the core	: 1"	: 2"	: 8"
Epithermal flux/thermal flux	: 0.022	: 0.0275	: 0.0050
Fast flux/thermal flux	: 0.68×10^{-3}	: 5.1×10^{-3}	: 0.038×10^{-3}
Flux γ Mr/h (at 10 KW)	: 86	: 250	: 85
Current of outgoing neutrons n/cm ² .sec.	: 0.8×10^9	: 0.67×10^9	: 0.77×10^9

The measurements confirm the decrease in fast neutron intensity. The high intensity of the gamma, even tangentially, is due to the capture gamma of the collimator. The neutron current mentioned in table I would correspond to a flux of 10^{15} neutrons/cm².sec. at the nose of the tubes. This increase with regard to the prediction is probably due to the reflections on the collimator. A measurement at the end of the tubes gives a flux of 0.52×10^{15} neutrons/cm².sec.

1.6.2 For the HFIR reactor, a joint ANL-ORNL program on the quality of emitted beams has been carried out at Oak Ridge, reference (5). The leaking radiations of HFIR should not differ much from those of AARR. Consequently, the use of this information should be in common. Two

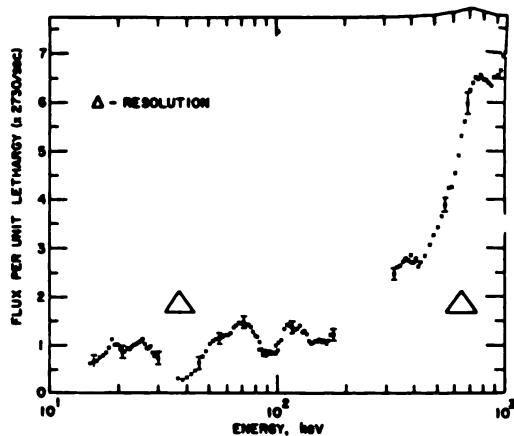


Figure 14 HFIR Axial-beam Spectrum

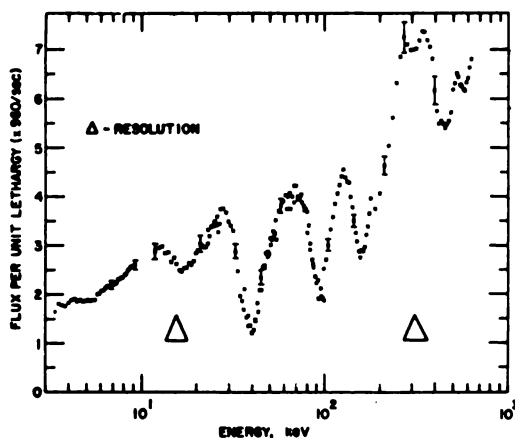


Figure 15 HFIR Tangent-beam Spectrum

horizontal HFIR tubes have been used, one radial (the HB-2) and the other one tangential (HB-3). The results of the measurements that are discussed in reference (5), are summarized in Fig. 14 and 15. They show that under 500 keV, the shapes of leaking spectra are not very different from each other. A rapid increase in the component of fast neutrons above the 500 keV can be viewed in the radial tube, but not in the tangential.

1.7 SPACE AVAILABLE IN THE REACTOR BUILDING FOR EXPERIMENTAL EQUIPMENT

In the HFBR, the distance between the plumb of the reactor and the wall of the reactor building is 23 m. In the GFHFR where the reactor is placed eccentrically in the reactor building, this distance is between 30 and 20; it is of 14.5 m in the UKHFBR, around 10 m in the HFIR and 15 m in the SM-2.

1.8 DURATION OF THE CYCLE

HFBR: The reactor operating cycle is nineteen days at full power, followed by a two days shut-down in order to change the position of half of the fuel elements, to replace the other half by new elements and to carry out necessary changes in the experimental equipment.

This duration is of twenty-six (H_2O) to thirty-eight (D_2O) days for the GFHFR, of twelve days for the UKHFBR and fourteen days for the HFIR.

At the SRHFD, the first five charges have worked six to eight days, the heaviest charges for nine to ten days. At SM-2, three or four fuel elements are changed every five days, without removal of the

reactor lid; the general change is carried out every twenty-five days with a stoppage of about one week. It is expected that AARR will have a cycle lasting ninety days.

1.9 ANNUAL EXPENSES OF THESE REACTORS

The prices of fuel cycles of the high flux reactors become very high in the region of 10^{15} neutrons/cm².sec. As an example, the annual expenditures of HFIR are given hereafter, reference (5).

ESTIMATED ANNUAL OPERATING EXPENSES OF HFIR (Includes ORNL overhead, etc.)

Operating Personnel	\$ 425,000
Technical Services	
Utilities (Electrical and Water)	\$ 197,000
Analytical Chemistry	10,000
Health Physics	30,000
Hot Cell Examinations	100,000
Liquid Waste Handling	60,000
Metals and Ceramics Division Support	95,000
Instrumentation and Controls Division	
Craft and Engineering Support	278,000
Mechanical Craft and Engineering Support	585,000
	<hr/> 1,355,000
Special Material	185,000
Control Plates (Material and Fabrication)	235,000
Beryllium and other Reactor Components	100,000
Fuel (does not include cost of uranium or reprocessing, 24 loadings)	1,800,000
	<hr/> \$ 4,100,000

1.10 PRESENT STATE OF THESE REACTORS

1.10.1 HFBR, Reference (8)

Since going critical in 1965, the HFBR has undergone all its tests, and particularly measurement periods on the beam tubes H¹, H² and H⁴ (Fig. 7), providing the intensities of the outgoing neutron beams, their spectra, their variation according to the point of the

nose of the beam tube under consideration, the fluxes at the beam tube noses, the heating in H^9 which is the position of the future cold source, etc. Note should be taken of the changes in intensity and temperature of the neutrons possibly due to the reflection of thermal neutrons from the walls of the collimator used. Measurements made on the interior extremity of the beam tubes show a flux of 5.2×10^{14} neutrons/cm².sec. at 40 MW which when compared to the figure of 1.0×10^{15} neutrons/cm².sec. deduced from the outgoing intensities leads to the conclusion that the number of neutrons undergoing a reflection in the collimator is nearly equal to the number of collimated neutrons. The isotropy of thermal flux has been verified at the end of H^2 . Finally, the outgoing intensity measurements checked with 3" and 10" collimation treated with tetradecanoic acid are in agreement with the flux at the extremity of the beam tubes.

1.10.2 GFHFR and UKHFBR

Both these reactors are now under study.

1.10.3 HFIR, Reference (5)

After criticality which took place in August 1965, the operations at low power lasted from January 29, 1966 to May 30, 1966 (influence of cuts in the currents, flux measurements, etc.). On May 31st, 1966, the power was increased to 20 MW, still with the first batch of fuel elements, until the 19th of June, 1966, the reactor being shut down because of the fuel being spent (2230 Mwd compared with 1800-1900 Mwd planned without targets). There was no unforeseen shut-down. On July 1st, 1966, a complete cycle at 75 MW was started again, to be followed by one at 90 MW, then at 100 MW.

1.10.4 SRHFD

The high flux campaign lasted from February to December 1965.

1.10.5 SM-2

SM-2 is working under full power since November 1962; it has been transformed since, in order to increase its power.

1.10.6 AARR

The reactor is being studied at the present time. Ground breaking is scheduled for 1967 and initial operation for 1970.

2. CHARACTERISTICS AND LIMITATIONS

2.1. CASE OF REACTORS SPECIALIZED IN OUTGOING BEAMS

We will consider here, as an example, only the direct limitations associated with a single aspect of the reactor.

In these high flux reactors with outgoing beams, the maximum

flux in the reflector increases with the specific power of the reactor. The specific power that can be obtained increases with the speed of the cooling water, with the surface of the heat transfer per volume unity (for example thin coolant channels), with the permissible temperatures and with the pressure. The speed of the cooling water, the exchange surface and the permissible temperatures depend more particularly on the fuel elements that are utilised. Let us study this particular problem in more detail.

The fuel elements of these reactors use the technology of aluminum-uranium plates clad with aluminum. The mechanical strength of the clad's alloy, Al 6061, imposes a limit on the temperature of the clad of about 216°, reference (10). The fuel materials have limitations of temperature depending on their burn-up; they are given in Fig. 16; reference (11). The units made up of these plates cannot resist to ultra high water speeds: in the elements with parallel plates, used for the HFBR, the speed is 10.7 m/sec. In the element with curved plates of the HFIR type, that has been envisaged for the GFHFR, this speed is 14 m/sec. In the element with annular plates that is planned for the UKHFBR, the speed amounts to 12 m/sec. The powers that can then be evacuated from these three reactors correspond at their hot spot to about 500 W/cm², 400 to 600 W/cm² and 400 W/cm² respectively. The corresponding pressures required are then 14 bars, 12 bars and 34 bars.

Which progress can be expected in these techniques ?

The aluminum of the clad should simultaneously possess suitable mechanical properties, reference (12) and a small corrosion. But this minimum of corrosion is sufficiently large for the aluminum oxide layer formed, especially contributes on the hot spot to limit the power density. GRIESS has shown that, as soon as the temperature of the solid surface-water interface increases above 170°C, the growth of the oxide film becomes significant. The low thermal conductivity of the aluminum oxide, besides the high heat flux, produces an appreciable temperature difference across the oxide film; this can attain about 50° under the conditions of GFHFR and the UKHFBR (the difference is about the same as for HFIR for which these results have been obtained). In order to reduce these limitations, the use of an aluminum alloy containing 1 % Fe, 1 % Ni, 1 % Mg is being studied concerning the GFHFR, for instance.

The increase of the limitations of temperature of the fuel should be possible by using the technology of UAl 3 (Fig. 16). The problems of satisfactory pile behaviour seem to have been resolved, reference (11). The problem remaining concerns the production of these

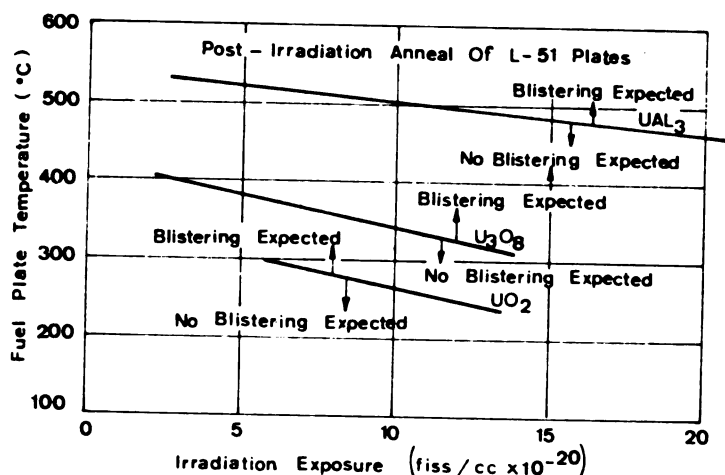


Figure 16 FAILURE-NO FAILURE LINES FOR VARIOUS FUEL PLATE CORE MATERIALS

(Extrait de IDO 17.154)

UAl 3 plates.

With all these expected improvements and, under the best conditions, starting from a heavy water cooled reactor, with 170° on the hot spot of the water-plate interface, i.e. 160° over a large surface, producing about 2.1×10^{15} neutrons/cm².sec., the clads' temperature would be increased up to 250°C, thus representing a progress of $\frac{250^\circ - 30^\circ}{160^\circ - 30^\circ} = 70$ per cent (the entrance temperature of the water into the core is 30°). This would yield $2.1 \times 1.7 = 3.5 \times 10^{15}$ neutrons/cm².sec. A supplementary gain on the speed could be expected. One supplementary m/s would bring 5.5 per cent beginning from 14 m/s. This, altogether, should lead us to the region of 4×10^{15} neutrons/cm².sec., i.e. a gain factor of two on the flux.

But let us leave this realist field for a futuristic one. We have seen that one limitation was due to the mechanical strength of the assemblage under hydraulic and thermal forces. Let us suppose that fundamental changes in technology lead to a much stronger fuel element and thus to higher speeds of, say, 30 m/sec. and to a maximum permissible temperature for the meat of 350°. The fuel element could be, for example, a block perforated with 3 mm diameter holes of 1.000 mm length arranged according to a lattice with a 4 mm spacing, with a total metal/water ratio of 1. One could extract 12 MW/l, the maximum heat flux being of about 1.800 W/cm², the up-stream pressure of 65 bars and the pressure-drop of 35 bars (above, the speeds would be limited by erosion). These 12 MW/l that could be extracted would make it possible to gain a factor

of four on the flux, based on the initial 2.1×10^{15} neutrons/cm².sec.; this would lead to the region of the 8×10^{15} neutrons/cm².sec.

These increased limitations would lead to new optimisations. No doubt, new problems connected with the behaviour of materials under strong irradiation, with transmutations, etc. would be encountered. Furthermore, an increase in the specific power, other things being equal, would decrease the length of the cycle. A new increase in the latter involving an increase of the maximum technological burn-up could be obtained by increasing the uranium charge by unit volume. This, in turn, requiring the increase of another technological limit. For the same load, this would necessitate an increase in the volume in order to obtain the necessary reactivity. This again, in turn, would decrease the useful flux/power ratio of the reactor.

To conclude, we can say that the beam emitting reactors make use of a technology which is close to its limit, and in which all progress of the existing technology should make it possible to obtain an increase by a factor two from the useful flux of 2×10^{15} neutrons/cm².sec. found in the most powerful reactors. A further factor of two, making four in all, could only be obtained by actually modifying present technology.

2.2 HFIR. Reference (5)

Preliminary research has been carried out with a view to examining the possibility of obtaining fluxes much higher than 10^{16} neutrons/cm².sec. in the flux-trap of an ultimate HFIR. Work by R. D. CHEVERTON has shown that an increase by a factor five of the specific power is possible, this thus giving 10 MW/l. It is necessary in particular to nickel-plate the fuel element surface in order to decrease corrosion. With only a small increase in size and using heavy water as coolant, it appears that a non-perturbed thermal flux of 3×10^{16} neutrons/cm².sec. can be obtained in the central part with a power of about 750 MW. For this, it is necessary that important improvements are made in the appropriate technology.

In addition to consideration of this "ultimate" HFIR type reactor, Mr CHEVERTON also considered the question of whether or not one could build an HFIR-type reactor to produce an unperturbed thermal neutron flux of 1×10^{16} , using only a modest extension of current technology. It was concluded that such a system could be built and that only a modest development effort would be required.

2.3 SRHFB. Reference (6)

With modifications in the design of the fuel elements, it

should be possible to develop charges giving a maximum thermal neutron flux of above 10^{16} neutrons/cm².sec. without increasing the heat fluxes.

3. PROSPECTS AND CONCLUSION

The reactors designed for producing neutron beams are based on the work carried out at Brookhaven which has defined the basic principles: undermoderated neutron sources, sources which are moderately absorbing in a heavy water reflector, tangential beam tubes, etc. At Harwell, studies have also been made on cores cooled by light water with beryllium reflectors and uranium-aluminum fuel elements. The flux/power yield was not much lower; at 100 MW a flux of 1.9×10^{15} neutrons/cm².sec. was obtained compared to 2.6×10^{15} neutrons/cm².sec. for the HFBR type. The signal/noise ratio, however, was much worse. For a core having stainless steel fuel elements with a charge of 43 kg of U²³⁵ and with heavy water as coolant and reflector, only 1.5×10^{15} neutrons/cm².sec. was obtained for the same power with a badly thermalized spectrum and a very uneven distribution of the power density in the core. Solutions exist to these problems; they have been studied by ANL.

GFK and CEA have compared the use as coolant of heavy water and light water, the reflector being heavy water in both cases. The first one has slightly better neutron characteristics than the second.

The improvements made in the field of the technology of fuel elements made up of aluminum-uranium plates, should make it possible to increase the maximum flux of thermal neutrons from the present 2.1×10^{15} neutrons/cm².sec. to 4×10^{15} neutrons/cm².sec. Radical changes in fuel element technology would perhaps make it possible to gain a supplementary factor of two and would lead to 0.8×10^{16} neutrons/cm².sec.

As to the reactors designed for irradiation purposes, the work carried out at Oak Ridge, based on a small-size reactor where the thermal high flux is obtained in a small flux-trap, has made it possible to obtain 2.0×10^{15} neutrons/cm².sec. perturbed and to study a considerable amount of problems arising in many other high flux reactors (fuel plates, corrosion, assembly fuel, etc.). The improvements in fuel elements should make it possible to reach 3×10^{16} neutrons/cm².sec in the central part, at 750 MW. 10^{16} neutrons/cm².sec. can be reached using only a small extension of technology.

Starting directly from this power level and using a very large size, Savannah River was able to develop directly a reactor load functioning with a maximum flux of 4.8×10^{15} neutrons/cm².sec. and which

should be able to reach 10^{16} neutrons/cm².sec. by modification of the fuel element.

All these realisations and these studies show that operating at the 10^{15} neutrons/cm².sec. level is already achieved or will very shortly be so. The improvements in the existing technology and particularly concerning fuel elements could perhaps make it possible to obtain significant increases of the fluxes.

4. ACKNOWLEDGMENT

I wish to express my gratitude to Mr. KOUTS who asked me to prepare this report, who suggested its scope and whose advice proved most invaluable.

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HFBR: A Source Reactor For Neutron Beams*

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1. Introduction

HFBR (High Flux Beam Reactor) is a 40 MW, heavy water cooled, moderated, and reflected reactor which has been built at Brookhaven National Laboratory to provide neutrons for basic research for physics and chemistry. The reactor was designed primarily for external neutron beam experiments, and the core, reflector, and beam holes are arranged to enhance the low energy neutron flux in the beams, while decreasing the fast background. Nine of the 16 experimental facilities are external beam facilities. Several of the beam holes are arranged so that two beams may be obtained from each hole. The remaining experimental facilities are for irradiation experiments.

The HFBR design has been described previously,⁽¹⁾ and that description will not be repeated here. Rather, this paper will concentrate upon the neutron beam facilities and the experiments now being carried out or planned for those facilities.

*This work was performed under the auspices of the U. S. Atomic Energy Commission.

II. Description of the Beam Facilities

A. Beam Holes: The main experimental facilities in the HFBR are the horizontal beam holes for external neutron beam experiments. There are nine such beams, with the characteristics of the various holes adjusted to suit the particular types of experiments to be done at each hole. The arrangement of the beam tubes in the reactor vessel is shown in Figure 1. This figure also contains a table listing all of the experimental facilities, their locations in the vessel, and the internal dimensions in the high neutron flux region.

The typical arrangement of the beam facilities in the biological shield is shown in Figure 2. Detailed plan views of several of the beam tubes are shown in Figures 3 and 4. Typical construction details for the beam tube and shutter case assembly are shown in Figure 5. Construction details of the standard beam plug are shown in Figure 6 and of the shutter in Figure 7.

Of the nine beam facilities, five are for thermal neutron beams. These are the facilities numbered H-3, H-4, H-5, H-6, and H-7. These tubes are arranged so that they do not "look" at the reactor core, and are sufficiently distant from the edge of the core so that the fast neutron background is greatly reduced and the predominant neutron energy in the beams is that of the thermal group.⁽²⁾ Two of the thermal beams, H-4 and H-6, are provided with large diameter liner sections in the biological shield to permit two neutron beams to be extracted from a single thimble in the reactor vessel. These facilities are called "dual thermal beams" in the table of Figure 1.

The beam facilities H-1 and H-8 are tangential beam tubes of

standard dimensions, placed nearer the reactor core than the thermal tubes to increase the epithermal component of the emerging neutron beam. The facility H-1 is located quite close to a corner of the core and will have the harder energy spectrum of the two. The facility H-2 is a radial beam tube which extends to within a few inches of the edge of the reactor core. It will be used in conjunction with a long flight path fast chopper apparatus.

The facility H-9 is a large diameter thimble in the reactor vessel, 12 in. I.D., which may be provided with a refrigerated moderator for the production of an enhanced cold neutron beam. The initial experiments to be done with the H-9 facility do not entail a refrigerated moderator, and simply use a neutron filter to obtain the existing low energy neutron component from the reflector of the reactor.

The portion of each beam tube facility which is cast into the biological shield consists of a port box, stepped tube, and shutter liner. The latter two components were factory assembled to insure alignment and to minimize field welding. The port boxes are inset approximately 1 ft from the shield face and vary in size from 23 in. square for H-1, H-3, H-5, H-7, and H-8, 35 in. square for H-4, H-6, and H-9, to 72 in. square for H-2. There are a pair of 3 in. pipes in the top and bottom of the port box which connect to upper and lower chases in the face of the biological shield. Through these pipes can be run power, gas, or vacuum lines from sources anywhere within the building.

The stepped tubes are accurately machined to a surface finish of 32 RMS for the larger diameter and 63 RMS for the small diameter.

The tolerance and concentricity of the bores are carefully controlled to maintain a .090 in. diametral clearance with the plug. The same careful attention is paid to the fit between the lower portion of the shutter and its casing. Here a 9/64 in. maximum clearance is maintained all around. The stepped tubes and shutter casing are fabricated from type 304 stainless steel plate. The containment flange, which does not come in contact with the plug, is a forged carbon steel ring.

There are drain fittings in the port box and shutter liner which connect to a manifold and thence to liquid-gas separators. If the effluent is water from a leaky shutter or shield plug, it will drain to a radioactive waste sump. Gas flow through these lines is vented through the separators to the building exhaust system.

B. Shutters: Each of the nine beam facilities is equipped with a shutter which can be moved vertically to close off the beam. A shutter consists of an upper heavy concrete shielding block and a lower water and steel filled box which is attached to the block. The location of the shutters and their dimensions are shown in Figures 3 and 4. An elevation of the shutter blocks in the biological shield is shown in Figure 2. Construction details of a typical shutter are shown in Figure 7. All of the shutters are 30 in. in length along the beam direction. The width varies from facility to facility, ranging from 7 in. for the standard beam holes to 20-1/4 in. for H-9. The shutter assemblies range in height from 6 ft-11 in. for beam tubes located above the core to 7 ft-11 in. for those below, and weigh from 3,000 lbs for H-2 and H-7 to 10,000 lbs for H-9.

The lower portion of the shutters, in the region of the neutron beam when the shutter is down, is composed of layers of 1% boron stainless steel in light water. The water is circulated past the steel plates in succession for cooling purposes. The steel plate assembly is enclosed in a water-tight type 304 stainless steel rectangular box the height of which varies from 11-1/2 in. for the smaller shutters to 19 in. for H-4, H-6, and H-9. Thermal neutrons are captured in a boron plate at the end of the water box facing the core to reduce the capture gamma ray burden on the shutter. The volume ratio of steel and water in the lower shutter is 60% steel and 40% water. The upper portion of the shutter block is filled with heavy concrete in a stainless steel box. Appropriate lifting lugs are provided at the top of the block for rigging. Each shutter is raised and lowered by means of an electrical motor operator of the rising stem type. These drive units are located in wells just beneath the pit seal floor. Gasketed cover plates in the seal floor permit access to the shutter motors for maintenance.

The shutter cooling water pipes extend upward through holes in a stationary shield plug mounted above the movable shutter block. The lifting screw also passes through a hole in this shield plug. The cooling pipes, which are of type 304 stainless steel, rise through packing glands in the seal floor. As the shutter moves up and down, the cooling water pipes slide up and down through the shaft seals. Several feet above the seal floor the pipes are connected to the cooling water manifolds through flexible stainless steel hoses. These are shown in Figure 2.

C. Beam Facility Shielding: For the horizontal beam tubes the shielding components of interest are the shutter, beam plug, and collimator, supplementary shielding in the port box and service chases, and the external shields around experimental equipment. The lower portions of the shutters, which intercept the beam in the down position, are constructed of boron-loaded stainless steel immersed in light water with a view to maximum attenuation of fast neutrons. The lower shutter box is closely fitted to the shutter casing and machined flat on the bottom to reduce leakage. The 30 in.-long shutter, composed of about 60% by volume of steel and 40% of water, has a fast neutron attenuation factor of about 9.0×10^5 and a core gamma reduction factor of approximately 2.5×10^4 . With the shutter closed and the reactor operating at full power, radiation levels are high at the port box. For the radial beam tube H-2 in particular, additional stopping material in the beam path is necessary to reduce the external beam to a level at which equipment may be serviced.

The standard stepped shield plug consists of an outer, or annular plug of steel and heavy concrete with appropriate water cooling coils. It has a central stepped hole which initially contains a solid plug of steel and heavy concrete. The combination of annular and solid plugs, together with the shutter in the closed position, gives a total attenuation in the beam tube which is very nearly that of the bulk concrete shield. The solid plug is replaced by a collimator assembly designed to suit a particular experiment. Figure 6 shows the standard annular plug for holes H-4, H-6, and H-9.

Many of the plugs for experiments differ from the design shown in Figure 6. In general, the experimental plug designs include some remotely controlled means of shutting off the beam in the plug itself. In some cases this is accomplished by rotating a barrel-type shutter within the plug and in other cases by flooding the collimator hole with light water. Without these additional shutters very brief access to external apparatus adjacent to the emerging beam is possible immediately after shutdown and unlimited access is possible 8 to 12 hours later with the main shutter closed. With the auxiliary shutter arrangement, complete access to the area in front of the beam hole is possible soon after shutdown.

The supplementary shielding in the port box area is a function of individual experimental setups. Where possible, experimenters stack shielding blocks into those regions of the port box which are not otherwise occupied by apparatus. The principal function of the port box is to provide a recess in the shield face into which the external shielding array may be stepped, to reduce radiation leakage along the crevice at the face of the biological shield. An arrangement of recessed horizontal chases and connecting vertical chases has been provided in the face of the biological shield to accommodate service lines. Light and heavy water manifolds run in the upper chase.

Even with small apertures in the plugs for the emerging beams, and with small-angle collimation, the radiation level of the beams outside the biological shield is very high. The external experimental apparatus must, therefore, be heavily shielded. The require-

ments for each experiment are more or less unique, depending upon the beam aperture, the collimation, and any filtering which may be done in the beam. In general, external shield arrays amounting to 3 to 4 ft of heavy concrete are required around the experimental apparatus. The control of the radiation level in the experimental area is a matter of continuing survey work by the Health Physics Group.

III. Current Use of the Reactor⁽³⁾

The four beam facilities with special geometry are H-1 and H-8 (the intermediate beam facilities), H-2 (the radial beam hole), and H-9 (the cold neutron facility). The beam tube ports through the biological shield at H-4 and H-6 are oversized. This feature permits simultaneous extraction of two divergent neutron beams from each. The remaining three beam facilities (H-3, H-5, H-7) are essentially identical in design, although small differences in the distribution of structural members in the reflector near the point of neutron extraction may lead to small differences in the characteristics of the emergent neutron beams.

In principle, then, the reactor can supply neutron beams to eleven simultaneous experiments. In actual fact, the numbers of beams and experiments is much larger than this number. Several of the beam ports supply multiple neutron beams, and the use of satellite experiments is increasing. It appears that the true limit on the number of simultaneous experiments will be set by the availability of space on the experimental floor, to permit separate blocks of apparatus to function with adequate freedom

from mutual interference. It seems at this time that from twenty to thirty separate experiments can be accommodated.

The intermediate beam facility H-1 is to be used for experiments with oriented nuclei, and usually with polarized neutrons. These experiments are meant to determine the angular momenta of states of the compound nucleus, to measure spin-dependence of cross sections, and to measure nuclear hyperfine interactions in solids. Monochromatic beams of intermediate energy neutrons are required. The intermediate character of the beam facility between radial and tangential orientation is chosen to supply neutrons especially in the energy range of interest. Two neutron beams are extracted, one to the principal crystal spectrometer at the reactor face, and the other to a second spectrometer nine meters beyond. A third and parasitic spectrometer will use a portion of the beam from the second spectrometer. Choices of collimators are available by mechanical rotation of sections of the beam tube plug (this feature is common to diffraction experiments at the reactor). High angular resolution is needed for precise energy determination (from 20 seconds of arc to 8 minutes of arc). Samples are cooled to $.02^{\circ}\text{K}$ by successive stages of refrigeration by liquid nitrogen, liquid helium, and diamagnetic depolarization, and are oriented by a superconducting solenoid producing a field of 50,000 oersteds. This experiment is in many ways the most complex one planned for the HFBR. It is not yet in place, but installation has begun, and initial operation is expected this year.

The radial beam facility at H-2 supplies neutrons to the fast

chopper. The rotor of the chopper spins at speeds up to 15,000 rpm, providing at this speed pulses of 1 μ s or 5 μ s width, as desired. At present, this chopper is used with a 23 meter flight path, for measurements of capture gamma-ray spectra. With this combination of rotor speed and flight path, measurements with adequate resolution are possible to neutron energies of several hundred electron volts. Gamma rays are detected by large lithium-drifted germanium detectors, with a resolution of about 0.1% at 8 MeV. Emphasis is being given to the higher energy primary gamma rays.

Two neutron beams are to be extracted from the beam facility at H-3. One of these is then to be further divided in two, so that altogether three experiments can be supplied by this facility. The experiments are generally devoted to capture gamma-ray spectroscopy. The unsplit beam will supply a crystal monochromator, to be used in measurements of γ - γ cascades following neutron capture below about 15 ev. Half of the split beam will supply a double-focussing beta-spectrometer, to be used in the study of conversion electrons following neutron capture in heavy elements. Neutrons will be fed to the target via a totally reflecting neutron guide. The other half of the beam will supply a second neutron guide for general purpose use. It is expected that the basic spectrometer will be installed, and the experiment under way, by about April, 1967. Installation of the beta-spectrometer and use of the third beam are expected to follow after the use of the crystal spectrometer has settled down.

The experimental holes H-4 through H-8 are shared by groups in the Chemistry Department and the Physics Department, in a

variety of solid state experiments. Since H-4 and H-8 provide dual beams, these facilities support seven crystal diffraction machines in a variety of kinds of experiments involving both elastic and inelastic scattering of neutrons. Included are four diffractometers at H-4 and H-6, a powder sample diffraction apparatus at H-5, a triple-axis spectrometer at H-7, and a double-axis diffractometer with neutron polarization apparatus at H-8. Compatible designs of these seven spectrometers provides for ready interchangeability of experimental apparatus (such as samples, sample holders, cryostats, etc.), and therefore a high degree of flexibility in the experimental programs.

Most of these seven spectrometers are now in place and producing data.

The spectrometers at H-4 through H-8 are all connected to an SDS-920 computer with a 16,000 word internal memory, a 32,000 word drum memory, tape units, typewriters, paper tape inputs, etc. This central control center programs the use of all the spectrometers, according to predetermined strategy, and also stores and reduces all of the output data on a real time basis. This multiple mode operation is provided simultaneously for all the machines through use of interrupts. Interrupt mode usage also permits communication with the computer at any time, through a typewriter or another such input.

The cold neutron facility H-9 supplies neutrons to a phased three-rotor time-of-flight spectrometer. This consists of two horizontal-shafted choppers preceded by a vertical-shafted chopper.

The use is for determination of dispersion laws and correlation functions from low energy inelastic scattering. In time, the source is expected to be a mass of liquid hydrogen in the large diameter beam tube. The original use will, however, be carried out with a chilled beryllium filter, viewing the neutrons as they are emitted from the inner end of the beam tube.

An SDS-910 computer with various inputs and outputs and a magnetic tape storage is used on a shared time basis by the fast chopper facility at H-1 and the cold neutron time-of-flight experiment at H-9.

In summary, it can be reported that the initially planned experiments, are about 70% operational at this time, about eight months after the achieving of full reactor power, and the remainder of these experiments should be operating within six months. The possibility of further use of split beams and satellite spectrometers should in time more than double the number of beam experiments over those originally expected.

IV. Features of the Beams⁽⁴⁾

The characteristics of the radiation emergent from the beam tubes was investigated during the design critical experiments, and in fact the design of the beam facilities was based on these experiments.⁽²⁾ Further and more definitive experiments have been performed during the startup phase of the reactor.⁽⁴⁾ Some of the results of the latter set of experiments will be reported here.

The measurements of beam characteristics during startup were performed on H-1, H-2, and H-4, these being representative of all of the existing beam facilities except H-9, which supplies the

cold neutron experiments. The thermal neutron intensity was measured at the inner tip of the beam tubes, and it was also deduced from the intensities of beams transmitted by a standard collimator. The results were as follows, normalized to 40 MW.

<u>Facility</u>	<u>Measured Thermal Neutron Source Intensity (n/cm²-sec)</u>	<u>Deduced From Collimated Beam (n/cm²-sec)</u>
H-1	6.3×10^{14}	5.7×10^{14}
H-2		4.8×10^{14}
H-4		4.8×10^{14}

These results are believed to be correct to within about 7%, with the uncertainty arising about equally from the power normalization and from the details of the flux measurements. The results show the effect of local depression of the neutron flux near the beam tube tips, since the peak value of the flux in the reflector was expected to be 7×10^{-4} -n/cm²-sec, uncorrected for the local perturbation.

The difference between the directly measured source flux and that deduced from the intensity of a collimated beam has not been explained. An interesting related observation was that the neutron flux in the beam tube near the inner tip is essentially isotropic.

The absolute epithermal neutron intensity in collimated beams was also measured by resonance detector activation, at five energies between 1 ev and 350 ev. The results are shown in Figure 8. It is seen that suppression of the epithermal intensity in the beam is indeed provided by the tangent orientation of H-4, while

the epithermal flux from the intermediately - oriented beam tube H-1 does in fact approach that from the radially - oriented tube H-2.

Fast neutron intensities in the collimated beams were measured by threshold detector methods. The measured relative intensities of collimated fast neutrons ($E > 335$ kev) in the three beams from H-2, H-1, and H-4 were observed to be in the ratios 1:0.136:0.0080. The ratios of fast to thermal neutrons in the three beams were found to be in the proportions 1:0.112:0.0082. These results are in reasonable agreement with those found in the critical experiments.

Measurement of the gamma ray intensities transmitted by collimators were not in agreement with the critical experiment results. The latter implied that the tangent orientation should reduce the collimated contaminant of gamma rays by a factor of several hundred. During the startup experiments, it was found that the gamma ray intensities in the three collimated beams were in the ratios 1:0.2:0.1 respectively, for the three facilities H-2, H-1, H-4. It is concluded that the reduction in gamma ray intensity in beams to be achieved by the choice of the tangent orientation is limited by neutron capture in the collimator. The collimators used in the startup experiments were made of steel, while those used in gamma ray flux determinations during the critical experiments were made of lead. The contrast of the results in the two cases indicates that considerable advantage may accompany the use of special collimator materials such as zir-

conium in experiments where low gamma ray background is desired.

V. Present Status of the HFBR

The HFBR was first brought critical at the end of October, 1965. The startup experiments and measurements of the physics parameters of the reactor were carried out in the succeeding months. The first full power run of the reactor was made early in February, 1966. The first normal operating cycle of the reactor was concluded in April, 1966.

Since April, the HFBR has been operating on a 40-day, 30 MW cycle. Vibration of tubes in the main heat exchangers has limited the secondary water flow rate on the exchanger shell sides to about half of the design value. The vibration occurs in the long, unsupported sections of the tubes in the U-bend end of the exchangers. The vibration amplitude at full design flow is not large, but it has seemed wise to eliminate any possibility of tube failures from this source by keeping the secondary flow rate below the threshold for measurable vibration. Piping changes are now in progress which will allow the exchanger shell sides to carry the design flow, but at half the velocity. These changes should eliminate the vibration problem. The piping changes involve converting the old secondary outlet line to an inlet, and cutting two new outlet lines into the exchanger shell at the U-bend end. These piping changes are currently being made during scheduled shutdowns of the 30 MW cycle.

Aside from the exchanger vibration problem, only the normal assortment of minor difficulties has been experienced in operating the plant. Most of these minor difficulties are in the function-

ing of the so-called "conventional" components of the system: pumps, valve operators, etc.

The start-up experiments have generally confirmed the neutron behavior of the reactor, as predicted in the design calculations and critical experiments. The control rods worths are slightly larger than the design values, and the equilibrium xenon reactivity is larger than had been calculated. The vessel and beam tube wall temperatures have been measured at 40 MW and are substantially lower than the design values. This last feature is a result of the very conservative viewpoint taken in the design calculations as to heating rates in these members, and to the heat transfer film coefficients on the vessel and beam tube surfaces. There is apparently a fair amount of turbulence in the reflector water, resulting in much higher film coefficients on the heated surfaces than had been expected.

VI. Power Escalation Capability of the HFBR

With the experience now in hand with the operation of the HFBR, it is possible to make some general remarks about the power escalation capability of the machine. It is clear that vessel and beam tube temperatures are not limiting, and that reactor power levels of 120 to 160 MW would be required to raise the vessel and beam tube wall temperatures to design values.

The primary system is operated at present at 200 psig cover gas pressure and 16,600 gpm flow. These could be increased to 250 psig cover gas pressure and about 19,500 gpm, giving a fuel element water velocity of 41 ft/sec and a core midplane saturation

temperature of 398°F. Under these conditions the HFBR could operate at 50 MW with the present heat exchangers and cooling towers. On very hot summer days, when the cooling tower water temperature approaches the 85°F design level, the core inlet water temperature would increase by about 10°F. The core hot spot would, however, be well below the saturation temperature.

An increase in power above 50 MW would require that cores composed entirely of fresh fuel elements be operated at reduced power for the first half of the cycle. Further, there would be some days each summer when the rise in the cooling tower water temperature would require a reduction in reactor power. The latter limitation could be eliminated by adding one more main heat exchanger to the system. With these qualifications it appears that the HFBR could be operated at 60 MW. Reactivity balance considerations might indicate that the fuel plates in the elements should be lengthened a few percent and the uranium concentration in the fuel alloy increased slightly, but these changes are well within the range of calculations and experiments which went into the design of the present element.

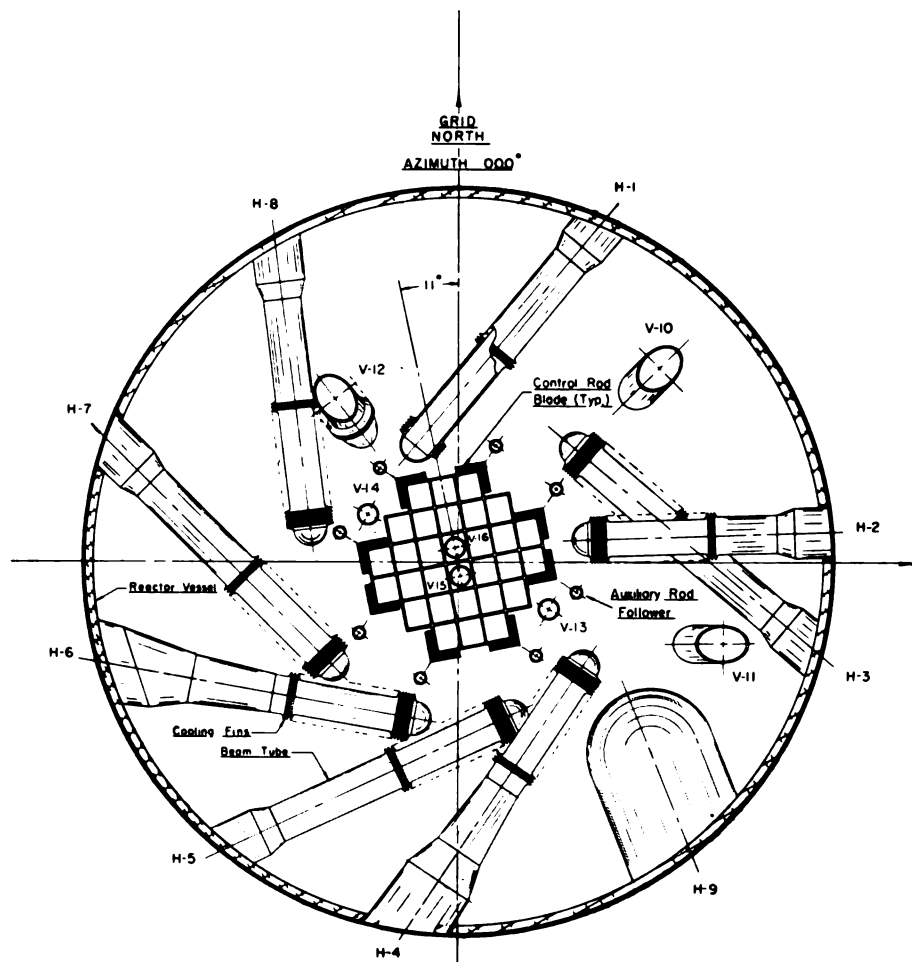
It is interesting to note that the allowance in the hot spot and burnout calculations which had been made for "dog-boning" have turned out to be unnecessary. The fuel fabricator for the HFBR elements has been able to make fuel plates with no detectable thickening of the fuel alloy at the ends of the plates.

In view of the recent examination of the ultimate power capability of the HFBR, it seems that the power level of approximately

60 MW required to supply a 10^{15} thermal neutron flux to the experiments can be achieved without substantial changes in the system.

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Experimental Facility	Azimuth	Elev. From Center	Radial Dist. From Center	Inside Diameter of Tube
H-1 Fast Neutron Beam	040°	-6"	13"	3.5"
H-2 Fast Chopper Beam	088°	+6"	12.9"	3.5"
H-3 Thermal Beam	130°	-6"	18"	3.5"
H-4 Dual Thermal Beam	214°	+6"	18"	3.5"
H-5 Thermal Beam	245°	-6"	18"	3.5"
H-6 Dual Thermal Beam	280°	on center	18"	3.5"
H-7 Thermal Beam	315°	+6"	18"	3.5"
H-8 Intermediate Beam	353°	on center	16.8"	3.5"
H-9 Thermal or Cold Beam	158°	on center	24.3"	12"
V-10 Thermal Irradiation Thimble	046°	-7.1" (end)	28"	3.5" (vessel); 1.87" (sample tube)
V-11 Thermal Irradiation Thimble	090°	-7.1" (end)	29.4"	3.5" (vessel); 1.87" (sample tube)
V-12 Thermal Irradiation Thimble	323°	-8.1" (end)	23.5"	3.5" (vessel); 1.87" (sample tube)
V-13 Core-edge Irradiation Thimble	139°	-6.1" (end)	11.3"	1.37" (sample tube)
V-14 Core-edge Irradiation Thimble	259°	-6.1" (end)	11.3"	1.37" (sample tube)
V-15 In-core Irradiation Thimble	319°	-13.1" (end)	1.6"	1.37" (sample tube)
V-16 In-core Irradiation Thimble	349°	-13.1" (end)	1.6"	1.37" (sample tube)

Figure 1 Diagram of the beam tube and irradiation thimble layout in the reactor vessel.

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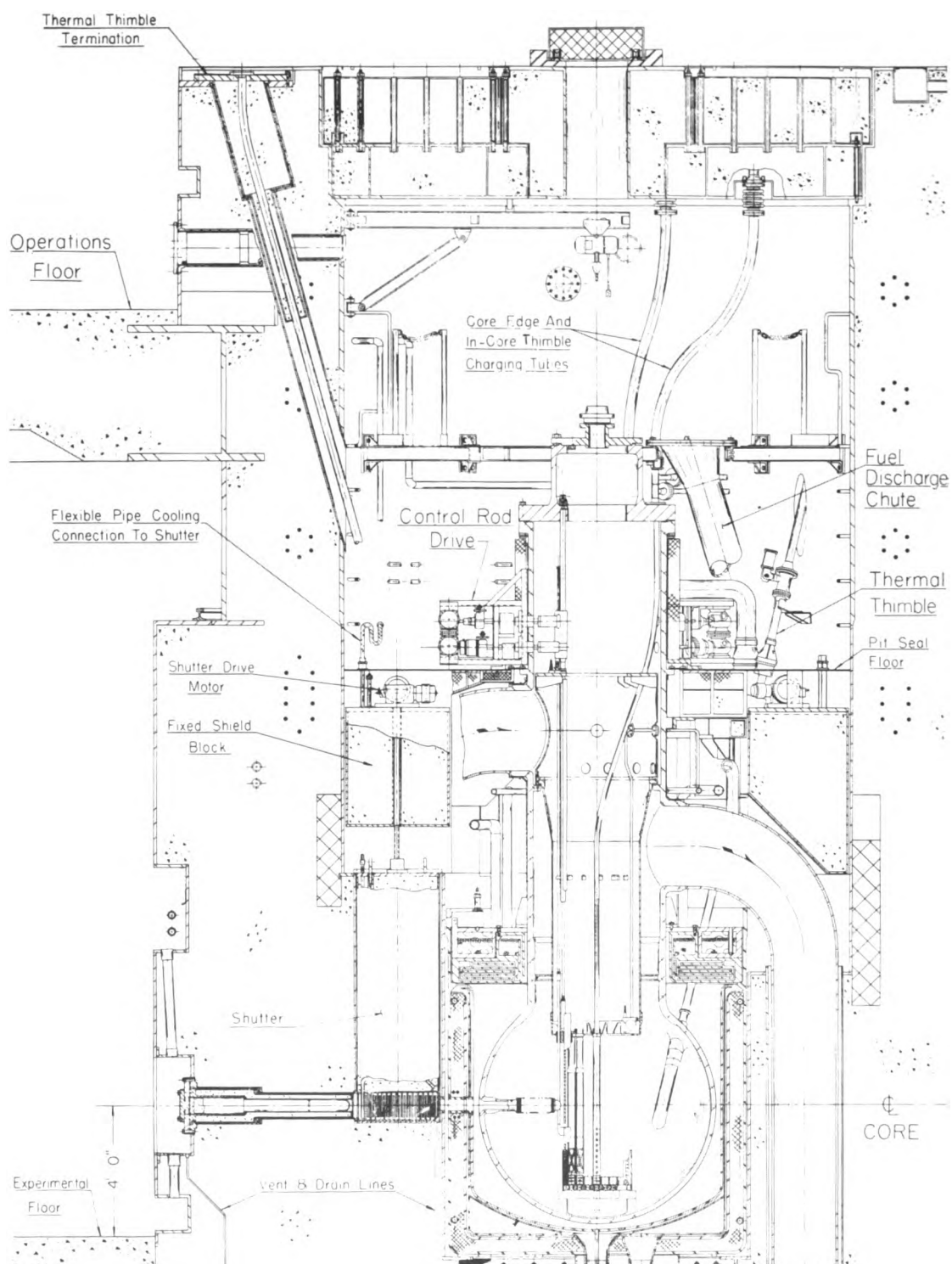
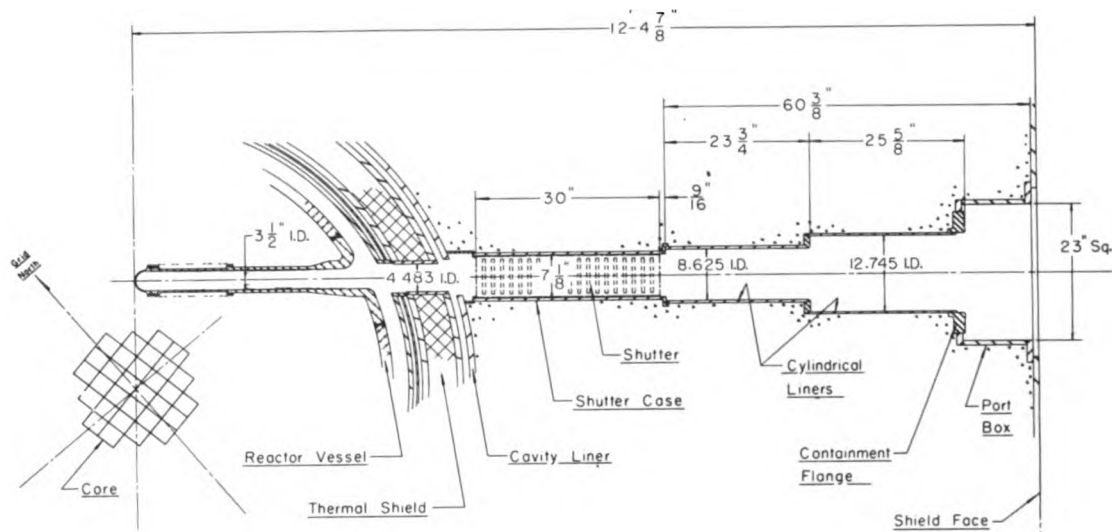
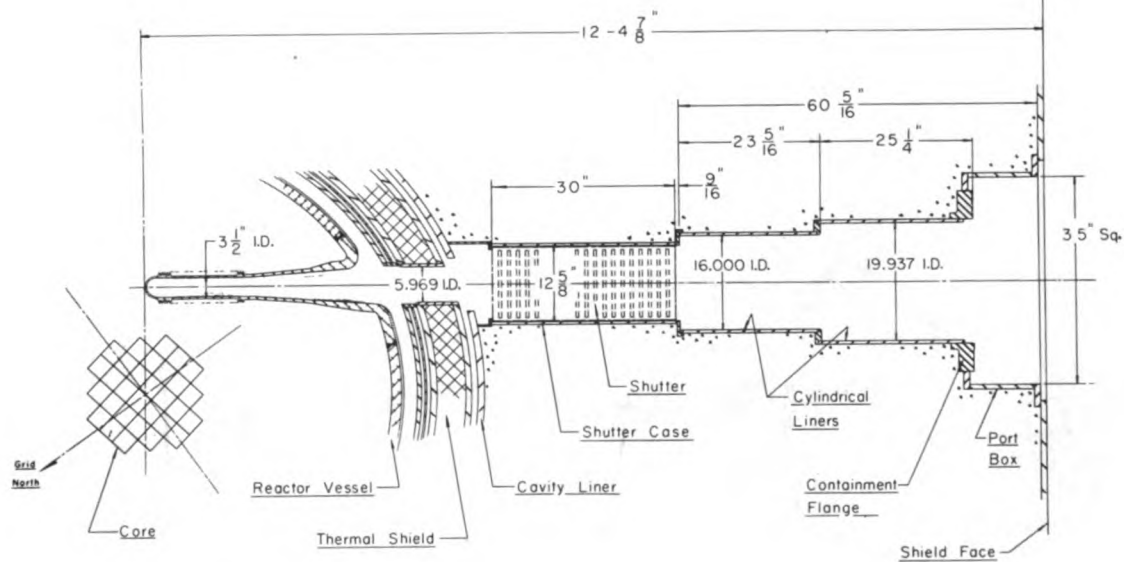


Figure 2 Elevation of the biological shield assembly showing the experimental facilities.



PLAN OF HFBR BEAM TUBE H-3
TYPICAL OF THERMAL TUBES



PLAN OF HFBR BEAM TUBE H-4
TYPICAL OF DUAL THERMAL TUBES

Figure 3 Plan views of beam tubes H-3 and H-4, from shield face to core.

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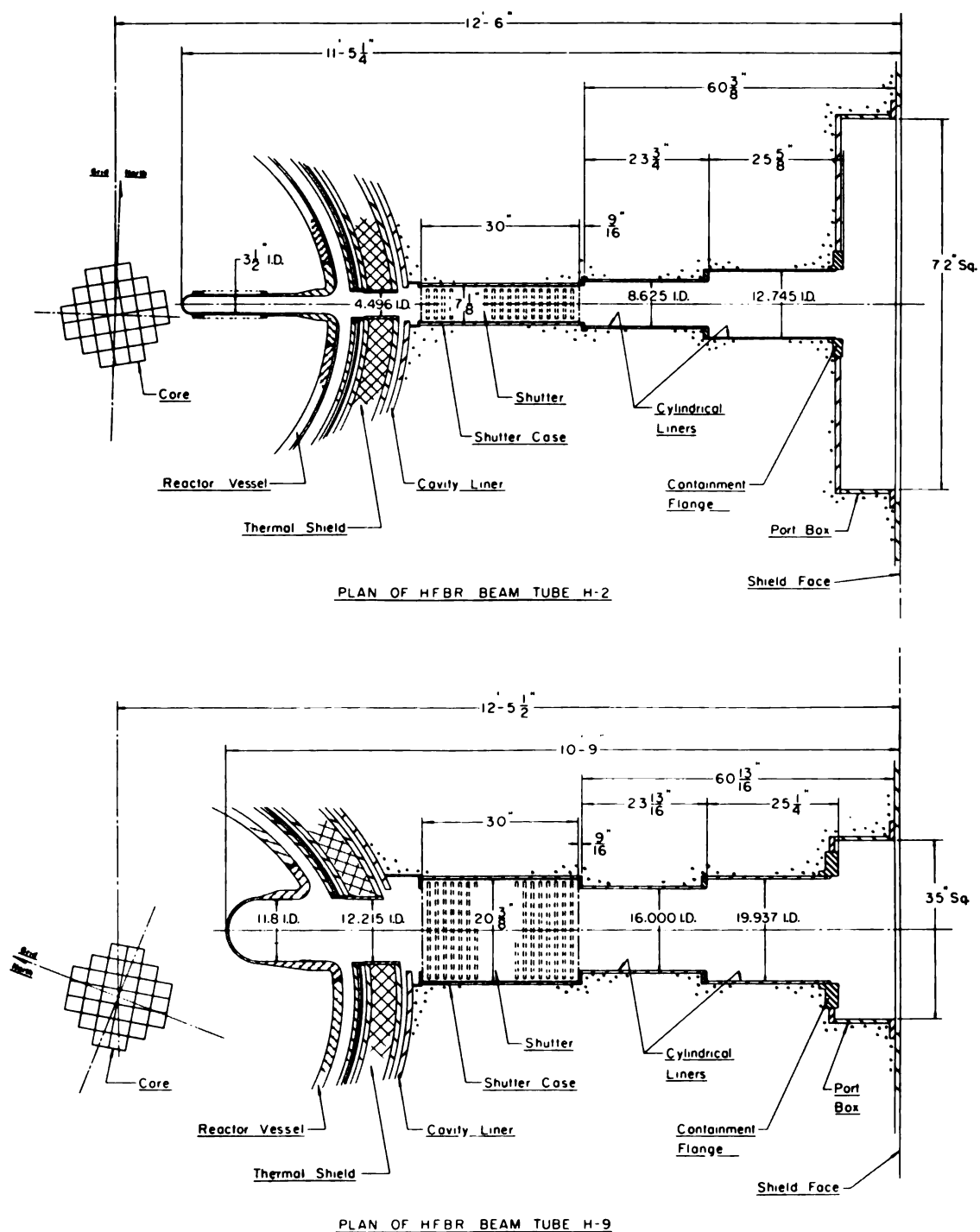


Figure 4 Plan views of beam tubes H-2 and H-9, from shield face to core.

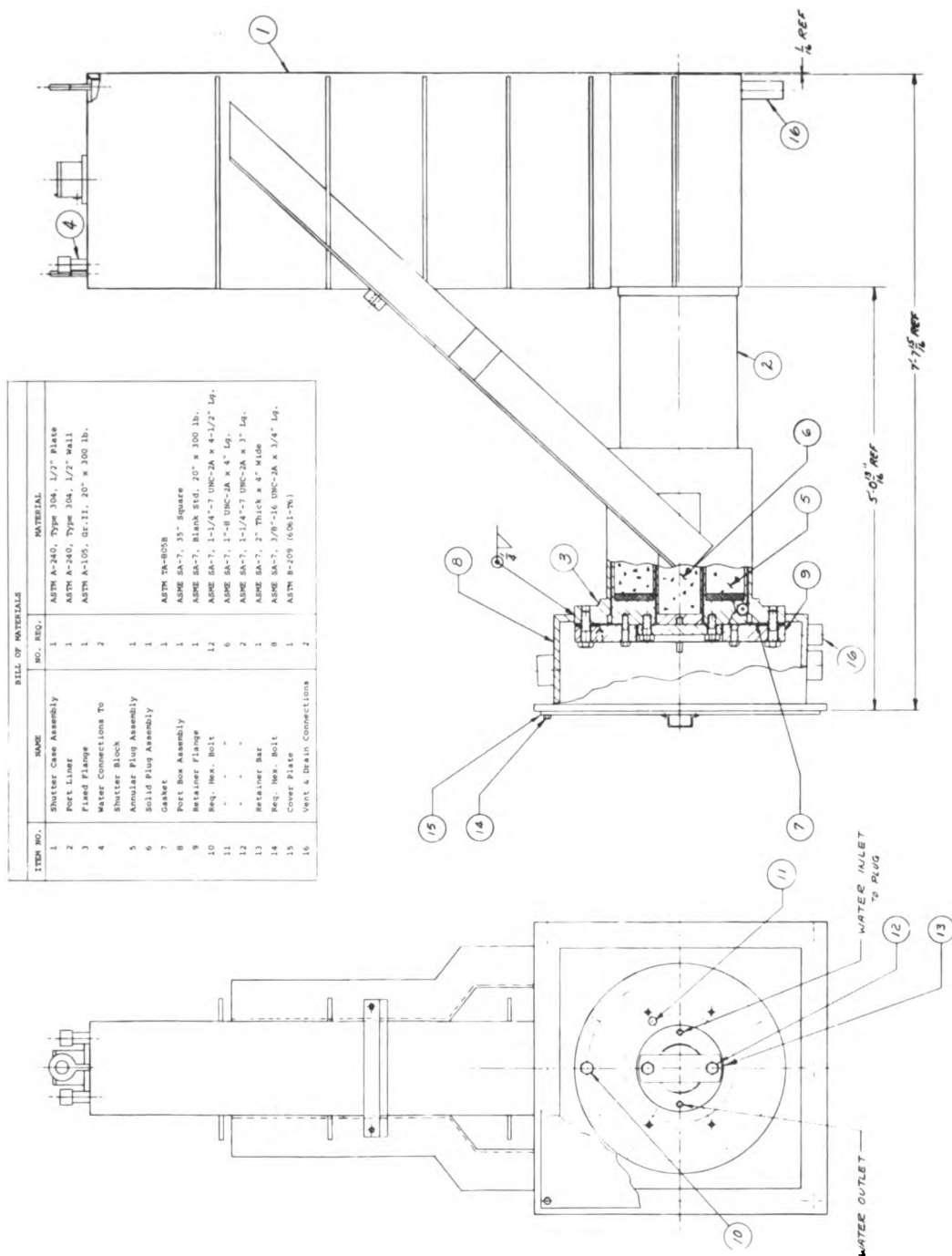
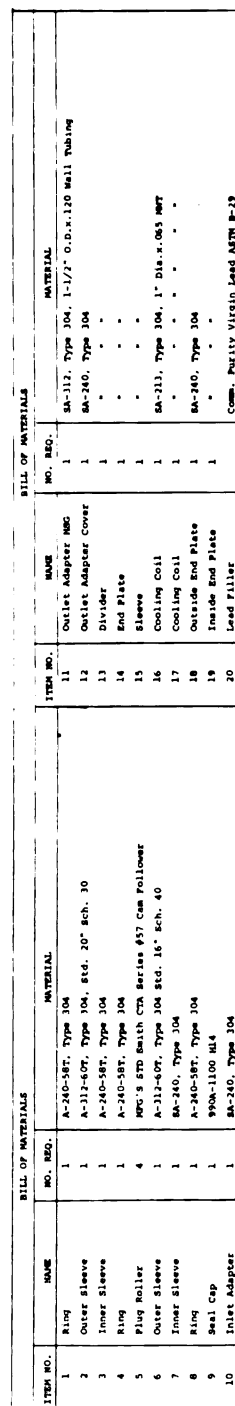


Figure 5 Beam tube and shutter case assembly for H-4 and H-6. The general construction is typical of all beam facilities.



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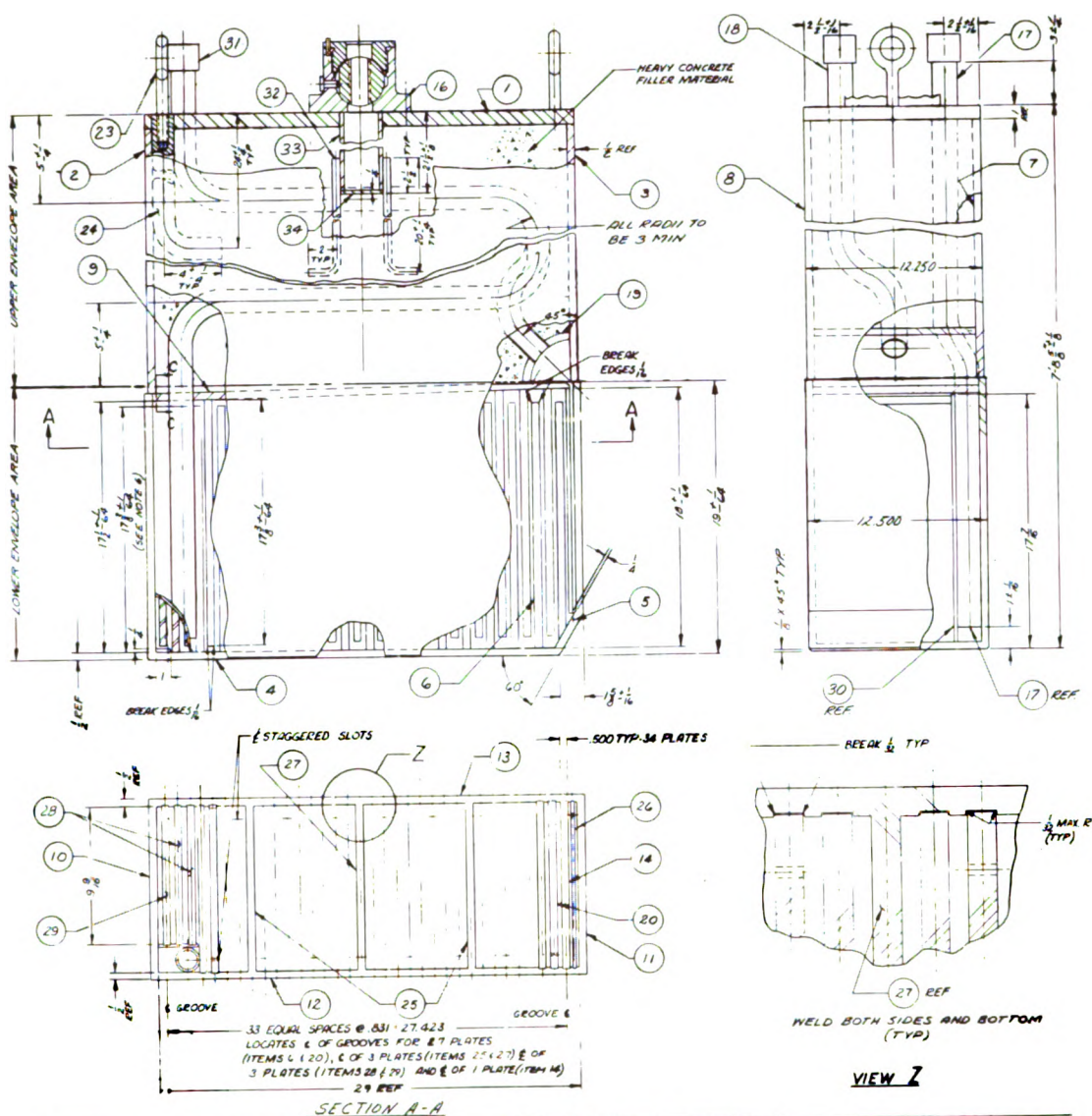


Figure 7 Detailed drawing of the shutter for H-6. The general construction is typical for all beam facilities.

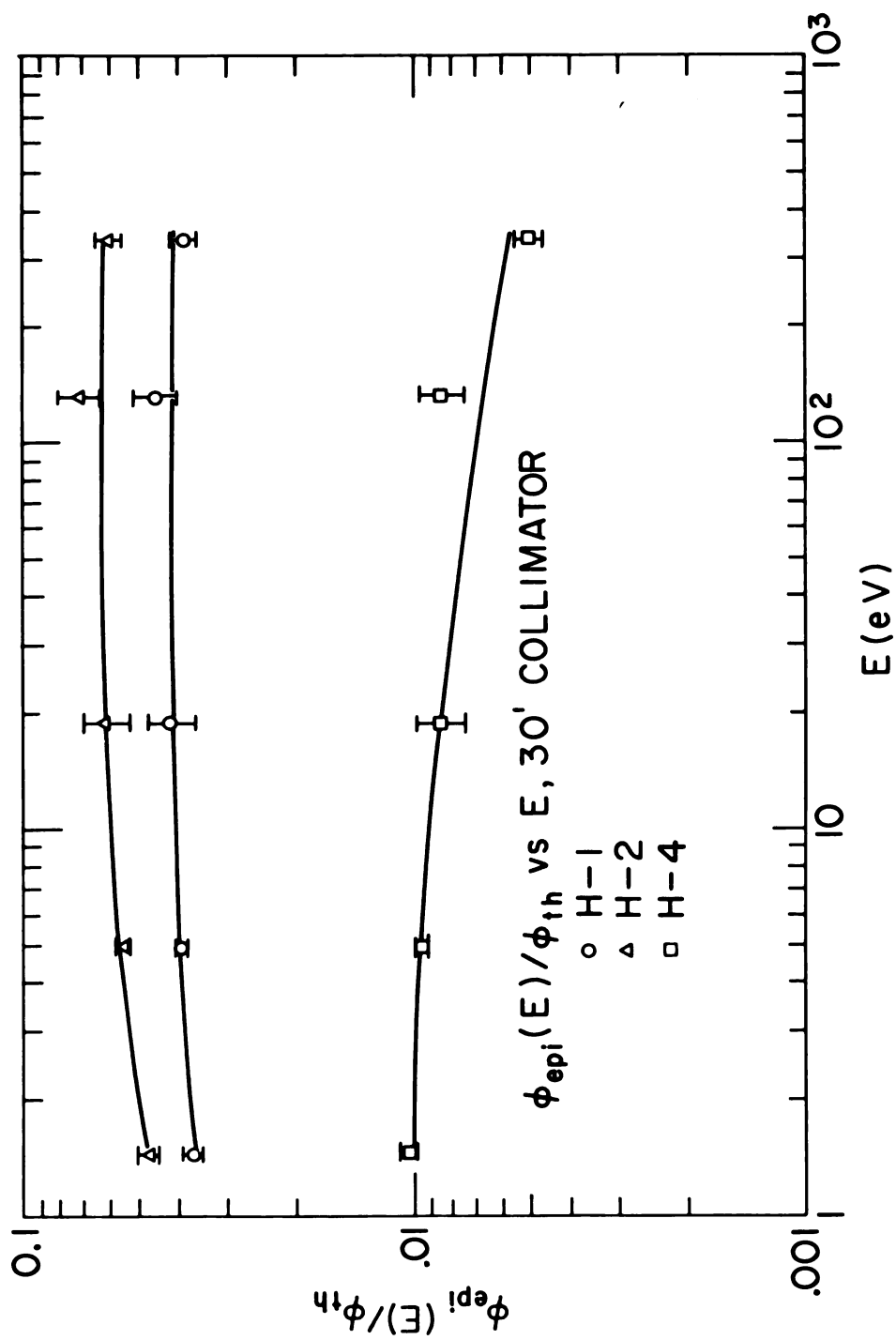


Figure 8 Epithermal vs. thermal flux from beam tubes.

THE OAK RIDGE HIGH FLUX ISOTOPE REACTOR*
DESIGN AND INITIAL OPERATION

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ABSTRACT: The High Flux Isotope Reactor (HFIR) was designed and constructed at the Oak Ridge National Laboratory (ORNL) primarily for the production of transuranium elements for use in the heavy-element research program of the United States [1]. The central target assembly will initially contain about 250 g of Pu^{242} which will be irradiated in the $\sim 2 \times 10^{15}$ n/cm².s flux of the HFIR to produce significant quantities (tens of milligrams) of the californium isotopes in addition to larger quantities of other heavy elements during the first year of irradiation.

Although the primary purpose of the HFIR is the production of transuranium elements in the high flux region, its usefulness is increased by several other experiment facilities which are located in, or terminate in, the reflector. These include four horizontal beam tubes, four slant access tubes, and 38 vertical holes of various sizes.

The HFIR is designed to operate at a maximum power level of 100 Mw. The fueled region is a cylinder having a volume of about 50 liters which leads to an average power density of about 2 Mw per liter. The 9.4 kg of U^{235} fuel is contained in curved, aluminum-clad plates 0.050 in. (1.27 mm) thick separated by 0.050 in. (1.27 mm) coolant channels. Ordinary water is used for coolant and moderator, and beryllium metal is used as a reflector.

The HFIR first achieved criticality on August 25, 1965. Following an extensive "zero-power" program the power level was increased to 20 Mw on January 29, 1966. Operation for a full fuel cycle at 75 Mw was completed on July 31, 1966, and it is anticipated that the design power level of 100 Mw will be achieved in early fall of this year.

The HFIR, including the building and auxiliaries, cost slightly under $\$15 \times 10^6$, exclusive of development cost. Development costs were approximately $\$6.7 \times 10^6$, of which about $\$2.5 \times 10^6$ was incurred in development of the fuel element. The annual operating cost, including the cost of fabricating the fuel elements, but not the cost of the fissionable material consumed or of reprocessing to recover the unburned fuel, will be about $\$3.8 \times 10^6$.

*Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

1. INTRODUCTION

The general features of the HFIR facility have been described previously [2,3]. Therefore, this paper will present only a summary description, taken primarily from reference [2], followed by information on the startup program, development, construction and operating costs, plans for utilization of the experimental facilities, and a few comments regarding the possibility of achieving a higher neutron flux with a reactor of this general type.

2. REACTOR CORE

Investigation of various reactor types and plutonium target designs indicated that a flux-trap type of reactor operating at ~100 Mw could produce the necessary thermal and fast neutron flux to meet the requirements of the heavy element research program. The desire to minimize development, capital, and operating costs led to selection of aluminum-clad fuel plates, normal water coolant and moderator, and a beryllium metal reflector, arranged in a core of cylindrical geometry.

The nuclear characteristics are summarized in Table I. A cross-section of the reactor in its pressure vessel is shown in Fig. 1.

Table I
General Nuclear Characteristics of the HFIR

Reactor power, Mw	100
Neutron fluxes, n/cm ² .s	
Maximum unperturbed thermal flux in island	5.5 x 10 ¹⁵
Average thermal flux in target (300 g of Pu ²⁴²)	2.0 x 10 ¹⁵
Average nonthermal flux in target	2.4 x 10 ¹⁵
Maximum nonthermal flux in fuel region	4.0 x 10 ¹⁵
Maximum unperturbed thermal flux in Be reflector	
Beginning of fuel cycle	1.1 x 10 ¹⁵
End of fuel cycle	1.6 x 10 ¹⁵
Prompt-neutron lifetime, μ s	
Beginning of cycle	35
End of cycle	70
Effective delayed neutron fraction	0.0071
Fuel loading, kg U ²³⁵	9.4
Length of typical fuel cycle, days	14

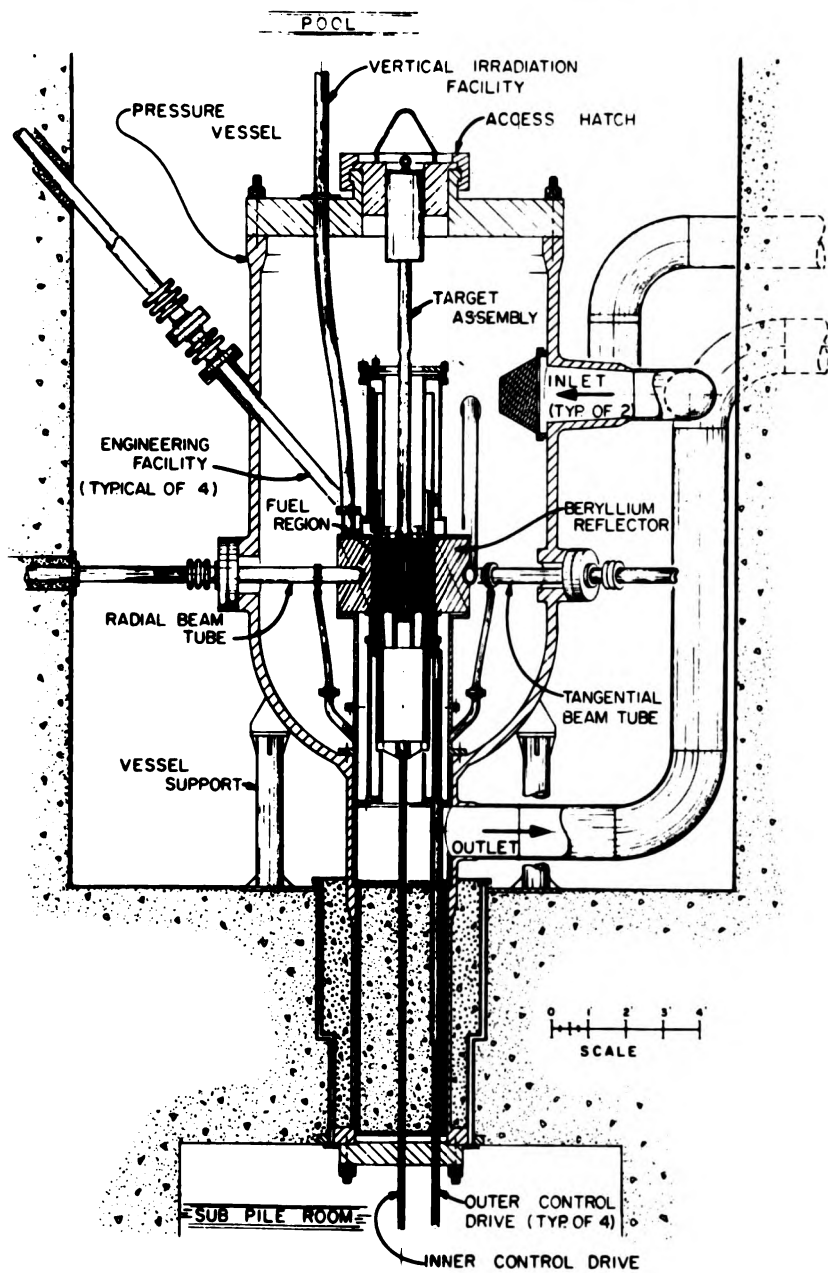


Fig. 1. High Flux Isotope Reactor
(Vertical Section)

The cylindrical HFIR core consists of four concentric regions. The central region, or flux-trap, contains normal water and is 5 in. (12.7 cm) in diameter, extending the length of the core. The fuel region located immediately outside the flux-trap consists of two cylindrical fuel elements containing U^{235} in vertical, curved plates. The fuel is surrounded by a cylindrical beryllium reflector, 12 in. (30.5 cm) thick. A normal water reflector of effectively infinite thickness surrounds the entire core. Control plates in the form of two poison-bearing concentric cylinders are located between the outer fuel element and the beryllium.

3. FLUX-TRAP PARAMETERS

Three major factors control the magnitude of the thermal neutron flux in the flux-trap: (1) the amount of neutron leakage from the fuel to the trap, (2) the slowing down and absorption characteristics of the trap moderator, and (3) the diameter of the trap-region. An optimum trap diameter exists because of the conflicting requirements for complete moderation (large trap-region) and for high neutron density (small volume). A moderator having the shortest slowing-down length provides the highest thermal neutron flux. Even though normal water has a relatively high thermal-neutron-absorption cross section, its slowing-down distance is sufficiently small to result in a higher flux in the trap-region than attainable with other moderators.

Neutron leakage from the fuel to the flux-trap is a function of core shape, core size, the metal-to-water ratio, power distribution, and the amount of parasitic absorption. Decreasing the core size and increasing the metal-to-water ratio increased the core leakage. Optimizing the length-to-diameter ratio of the fuel region increased to a maximum the total leakage per unit core height into the island.

The limitation on reduction of the core volume was associated with heat removal characteristics. A significant improvement in reactor performance was attained by incorporating features which reduced the ratio of maximum-to-average power density (q_{\max}/q_{ave}) to 1.45 (exclusive of hot-spot factors). These features include cylindrical core geometry, a symmetrical reflector control system, and radial variations in the fuel and burnable poison concentrations. By comparison with a uniform fuel distribution, the radial q_{\max}/q_{ave} was reduced from 3 to 1.2.

4. FUEL ELEMENTS

The compact fuel assembly of the HFIR and its dimensions are shown in Fig. 2.

The inner element, which contains 171 plates, is initially loaded with 2.6 kg of U^{235} and 2-4 g of B^{10} as burnable poison. The outer element contains 369 plates, initially loaded with 6.8 kg of U^{235} and no burnable poison. The individual plates are of a sandwich-type construction with a fuel-bearing cermet

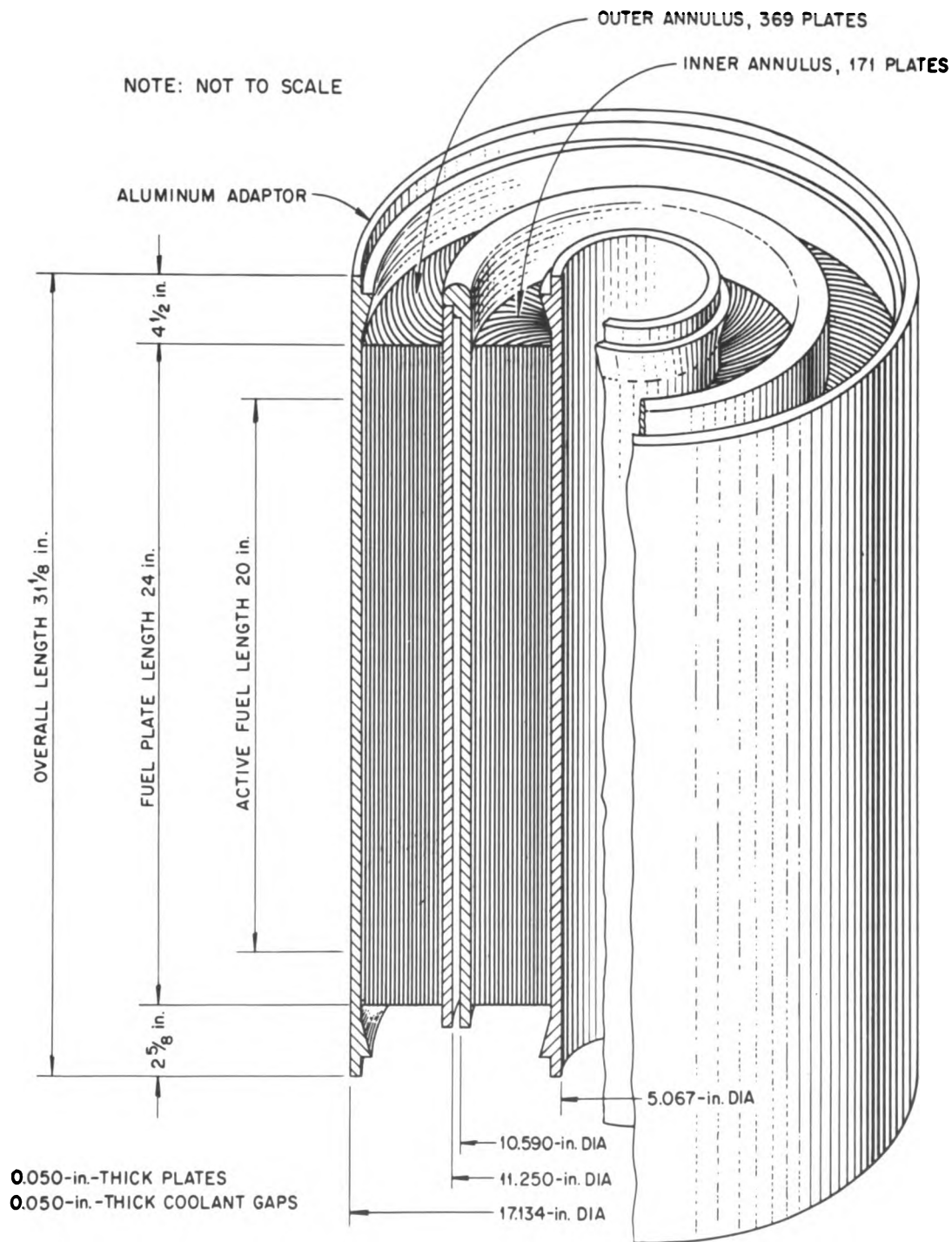


Fig. 2. HFIR Fuel Element

sealed between covers of 6061 aluminum. The cermet is a mixture of aluminum and U_3O_8 , 35% U_3O_8 by weight in the inner plates and 40% in the outer. The finished plates are 0.050 in. (1.27 mm) thick, including a minimum cladding thickness of 0.010 in. (0.25 mm). The plates have an involute curve which provides a cooling channel between plates of constant width, 0.050 in. (1.27 mm). The plates are welded to the cylindrical aluminum side plates.

The radial variation in the fuel and poison concentrations is obtained by varying the thickness of the cermet cores across the width of the fuel plate as illustrated in Fig. 3. The B^{10} , in the form of B_4C , is added to the aluminum filler pieces of the inner element fuel plates.

5. CONTROL PLATES

The control "rod" system of the HFIR was selected primarily for its ability to control reactivity adequately without introducing undesirable perturbations and asymmetries in the power distribution or in the neutron fluxes of the flux-trap. The system constitutes a reflector control system that regulates the flow of thermal and near-thermal neutrons from the beryllium reflector to the fuel region. As shown in Fig. 4, the narrow annulus between the fuel and the beryllium reflector contains two thin, concentric cylinders which are separated from each other and from the adjacent regions by narrow coolant gaps. The inner cylinder has a single drive rod and is used for both shim and regulation. The outer cylinder is divided into quadrants, each with its own drive rod and release mechanism; these four control plates are used for both shim and safety. During normal operation, the four shim-safety plates and the shim-regulating cylinder are moved in unison to avoid asymmetries in power distribution. When used in an emergency, the shim-safety plates are released separately, thus providing multiplicity for shutdown.

To hold variations in the longitudinal power distribution within acceptable limits and to prolong the neutron absorber life of the rods, the control cylinders are divided into three discrete longitudinal regions: a highly neutron-absorbing (black) region, a moderately absorbing (gray) region, and a comparatively poorly absorbing (white) region. By locating the black regions of the two control cylinders at opposite ends of the core (Fig. 4) and by moving the cylinders in opposite directions symmetrically, it is possible to maintain power distribution symmetry about the horizontal midplane of the core.

Materials selected for the control plates are a 33-volume percent Eu_2O_3 -Al dispersion clad with aluminum for the black region, a 40-volume percent Ta-Al dispersion clad with aluminum for the gray region, and solid aluminum for the white region.

6. BERYLLIUM REFLECTOR

Because of the high performance required of the HFIR and because of its small size, the choice of reflector was limited to beryllium or D_2O . Beryllium

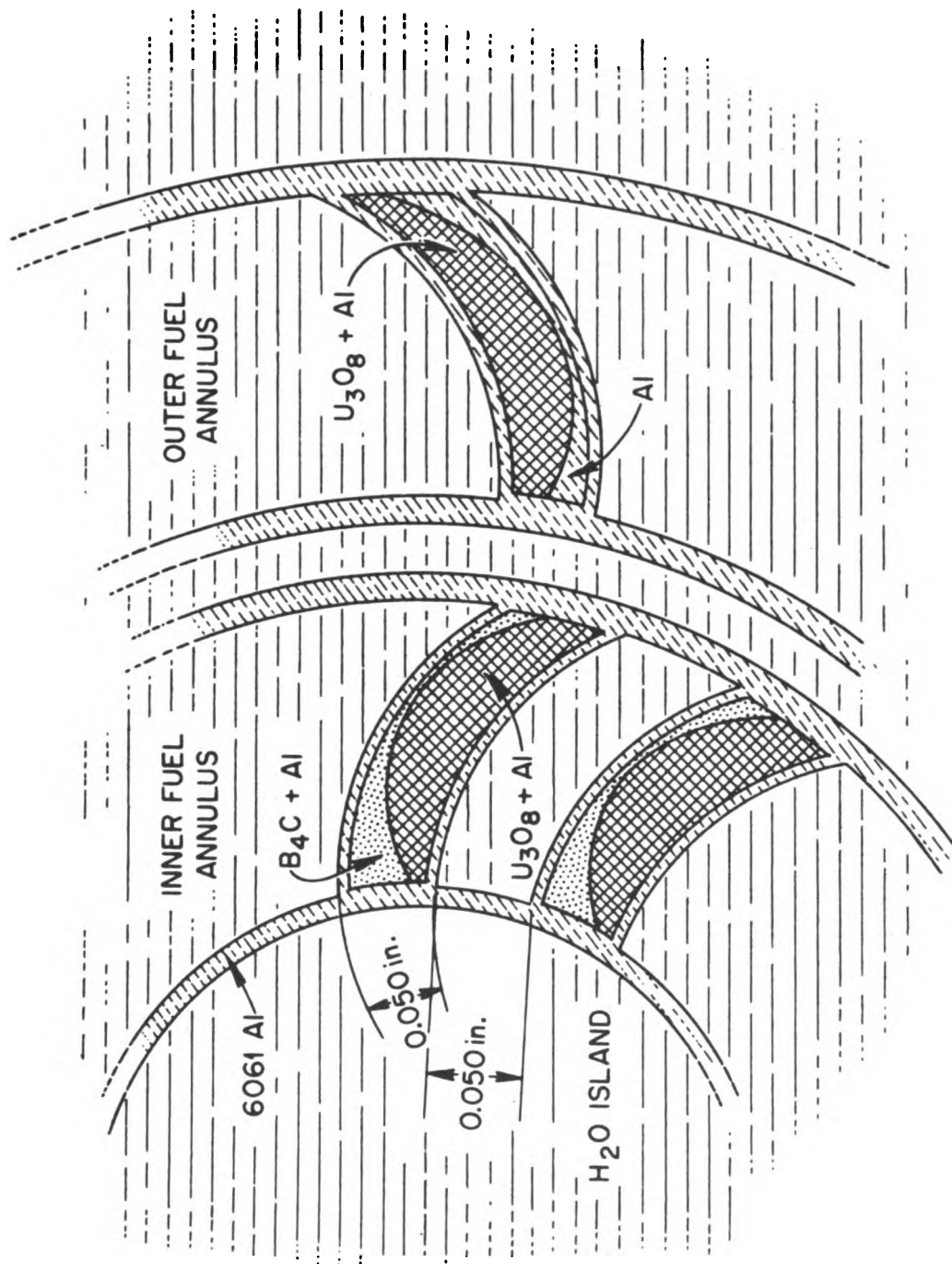


Fig. 3. Schematic Representation of Core Cross Section, Showing Fuel Contours

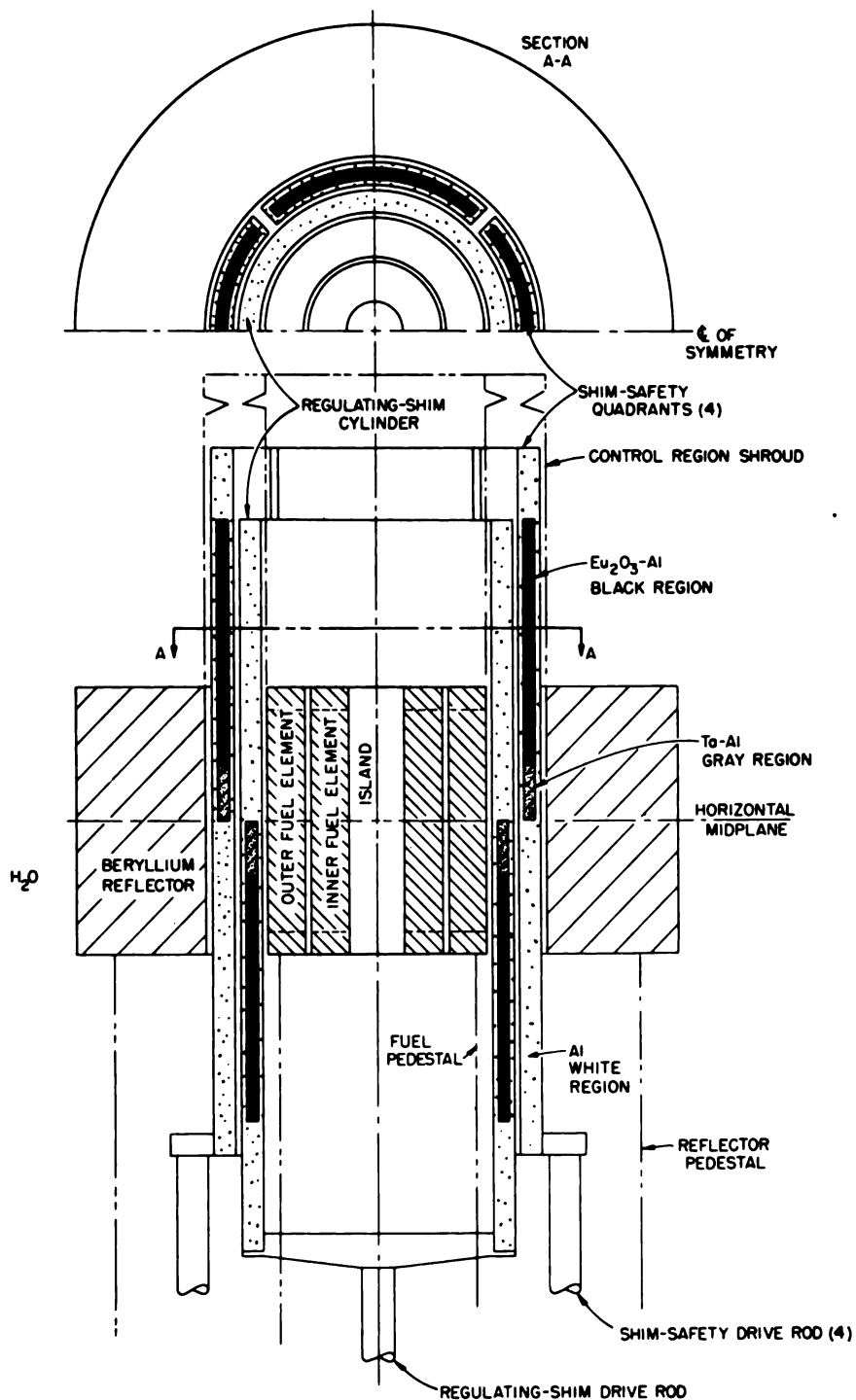


Fig. 4. Schematic Representation of Control Rod Arrangement in HFIR;
Rod Position Shown Approximate for Clean, Critical Core

was selected initially because of generally satisfactory, long-term experience with beryllium reflectors, the advanced state of beryllium fabrication, and the relatively complex auxiliaries necessary to cool D_2O . Although the magnitude of the deterioration of beryllium in the Materials Testing Reactor (MTR) did not become known until design of the HFIR was well along [4,5], the reflector had been designed as three concentric regions with provision for simple replacement of the inner region. The MTR experience indicated that exposure of the HFIR beryllium should be limited to $1-2 \times 10^{22}$ nvt of fast neutrons ($E > 1$ Mev) unless substantial mechanical distortion and/or cracking could be tolerated. Because the inner region of the beryllium would accumulate this exposure in a few months, revisions in the design were made to preclude operational difficulties which might arise from cracking or distortion of the beryllium before replacement.

7. HEAT TRANSFER AND CORROSION

Careful calculation and experimental verification of heat-transfer characteristics of the HFIR core were required because of the desire to extract as much power as practical from the smallest practical core. As indicated in Table II, the configuration that was selected leads to an average power density in the core of almost 2 Mw/l^{-1} , and an average fuel element heat flux of $7.7 \times 10^5 \text{ Btu hr}^{-1} \text{ ft}^{-2}$ ($58 \text{ cal s}^{-1} \text{ cm}^{-2}$).

A power level margin of, at least, 1.3 in heat transfer rate is provided over that corresponding to the onset of nucleate boiling at a hot spot. Calculations of this margin take into account the fuel plate tolerances, creep of the fuel plate during operation, fuel loading variations and non-bonding in the plates which may be below the sensitivity of detection. Measurements of heat transfer rates, burnout heat flux, plate deflection resulting from thermal and hydraulic forces and flow through the fuel assembly have been made to verify the calculations used in the heat transfer design.

The ability to use aluminum for fuel plate cladding and core structural material depends upon satisfactorily low fuel plate and interface temperatures, as well as adequate corrosion resistance to the water coolant. With respect to the latter, not only corrosion per se, but also the formation of aluminum oxide films on the fuel plate surfaces which would adversely affect heat-transfer rates, were of concern.

An experimental corrosion program established that aluminum, particularly type 6061, should be satisfactory under the anticipated conditions provided the acidity of the water coolant is adjusted to about $\text{pH} = 5$. Under these conditions, corrosion is expected to be satisfactorily low and film growth during the expected life of the fuel plates should be sufficiently low that acceptable fuel plate temperatures will be maintained.

The oxide buildup results in an appreciable increase in plate temperatures,

Table II
Heat Transfer and Removal

Total reactor power, Mw	100		
Power in fuel region, Mw	97.5		
Volume of active core, liters	50.6		
Heat transfer area	40	m ²	428.8 ft ²
Coolant velocity through core	15.5	m s ⁻¹	51 ft s ⁻¹
Inlet water temperature	49	°C	120 °F
Exit water temperature, average	73.3	°C	~164 °F
Coolant flow through fuel channels	~812	l s ⁻¹	~13,000 gpm
Coolant flow, total	~1000	l s ⁻¹	~16,000 gpm
Coolant inlet pressure	41	atm	~ 600 psig
Power density, Mw/l			
Average	1.9		
Hot spot	4.3		
Heat flux	<u>Cal s⁻¹ cm⁻²</u>		<u>Btu hr⁻¹ ft⁻²</u>
Average	58		7.7 x 10 ⁵
Hot spot	151		2.0 x 10 ⁶
Incipient boiling power level, Mw	>130		

Temperatures, °F (with 121°F inlet water temperature):

	<u>Start of Cycle</u>			<u>End of 15-day Cycle</u>		
	Fuel Average	Hot Streak Outlet	Hot Spot	Fuel Average	Hot Streak Outlet	Hot Spot
Fuel plate temp, °F	205	312	390	212	360	480
Metal-oxide interface temp, °F	197	303	371	204	351	461
Oxide-water interface temp, °F	197	303	371	197	303	377
Water outlet temp, °F	178	246	246	178	246	246

as evidenced by the hot-spot metal temperature (Table II) which increases by about 100°F (55°C) during the fuel cycle. Since plate strength is more closely related to plate average rather than hot-spot temperatures, it appears from these data that plate strengths will be adequate throughout the fuel cycle. While the available data on oxide buildup and irradiation damage [6] indicate that operation at 100 Mw should be possible, it is in this area that the major uncertainties re-

garding performance are present. The combined effects could possibly lead to the necessity to limit the power level and/or core lifetime until modifications to the fuel element design are made to counter the problems encountered.

As will be described in a later section, the approach-to-power phase of the program includes the operation of the reactor for a full fuel cycle (until shut-down is caused by fuel depletion) at each of several successively higher power levels to demonstrate fuel element behavior under conditions of approximately the same fuel burnup but at successively higher fuel plate temperatures.

8. TARGET ASSEMBLY

The target assembly, shown in Fig. 5, is located in the flux-trap region and contains the Pu^{242} target material. The thirty-one 3/8-in. (0.95-cm) OD aluminum tubes are filled with pellets made from a mixture of PuO_2 and aluminum powder. The tubes are capped and collapsed onto the pellets to provide good heat transfer. These are centered by fins integral with the tube inside shrouds which define the coolant channels. The shrouds are clamped securely in a framework which accurately positions the tubes and facilitates removal and maintenance operations.

For an initial charge of 300 g of Pu^{242} , which is calculated to be about optimum for maximum heavy isotope production, the heat generated in the target will be about 900 kw. This is removed by a flow of about 675 gal min^{-1} (42.6 ls^{-1}) around the target rods, and about 113 gal min^{-1} (7.1 ls^{-1}) between the target assembly and the inner fuel element.

9. EXPERIMENT FACILITIES

Although the primary purpose of the HFIR is the production of transuranium elements in the flux-trap region, its usefulness is increased by several other experiment facilities which are located in, or terminate in, the reflector. These include four horizontal beam tubes, four slant access tubes called "engineering facilities," and 38 vertical holes. Figure 6 shows the general arrangement of these facilities in the core region.

A hydraulic tube for irradiation of small samples in the target region is planned for future installation. This tube would replace the central target rod and would permit short irradiation in the very high flux region without interfering with operation of the reactor.

The four nominally 4-in. (10-cm) ID horizontal beam tube experiment facilities extend outward from the reactor core at the midplane. One beam tube extends radially from the reactor center line with its inner end penetrating the permanent reflector. Another tube extends tangentially from the core, offset approximately 10.5 in. (26.7 cm) from the reactor center line. It also penetrates the permanent reflector. The remaining two tubes are aligned on a tangential line approximately

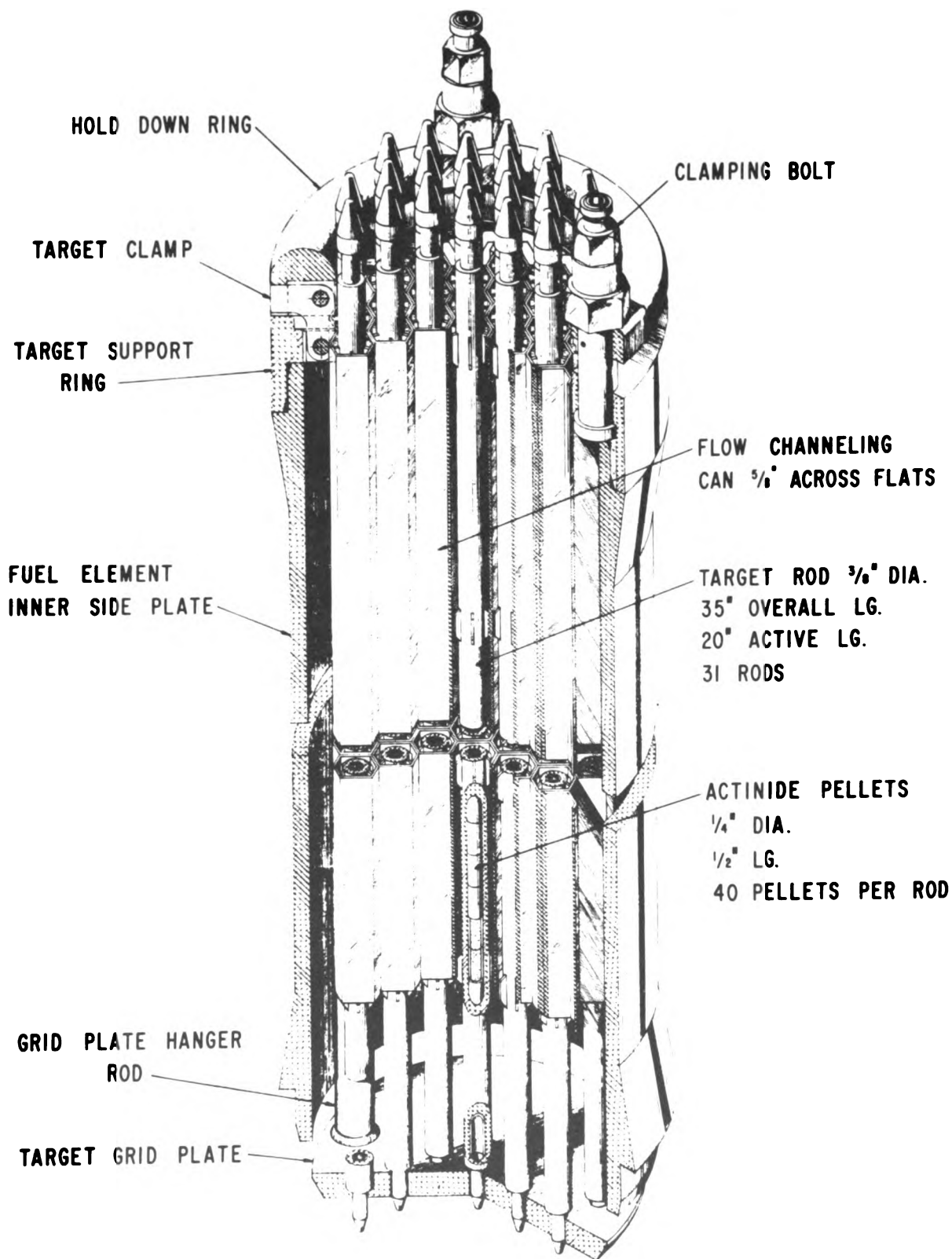


Fig. 5. HFIR Target Assembly

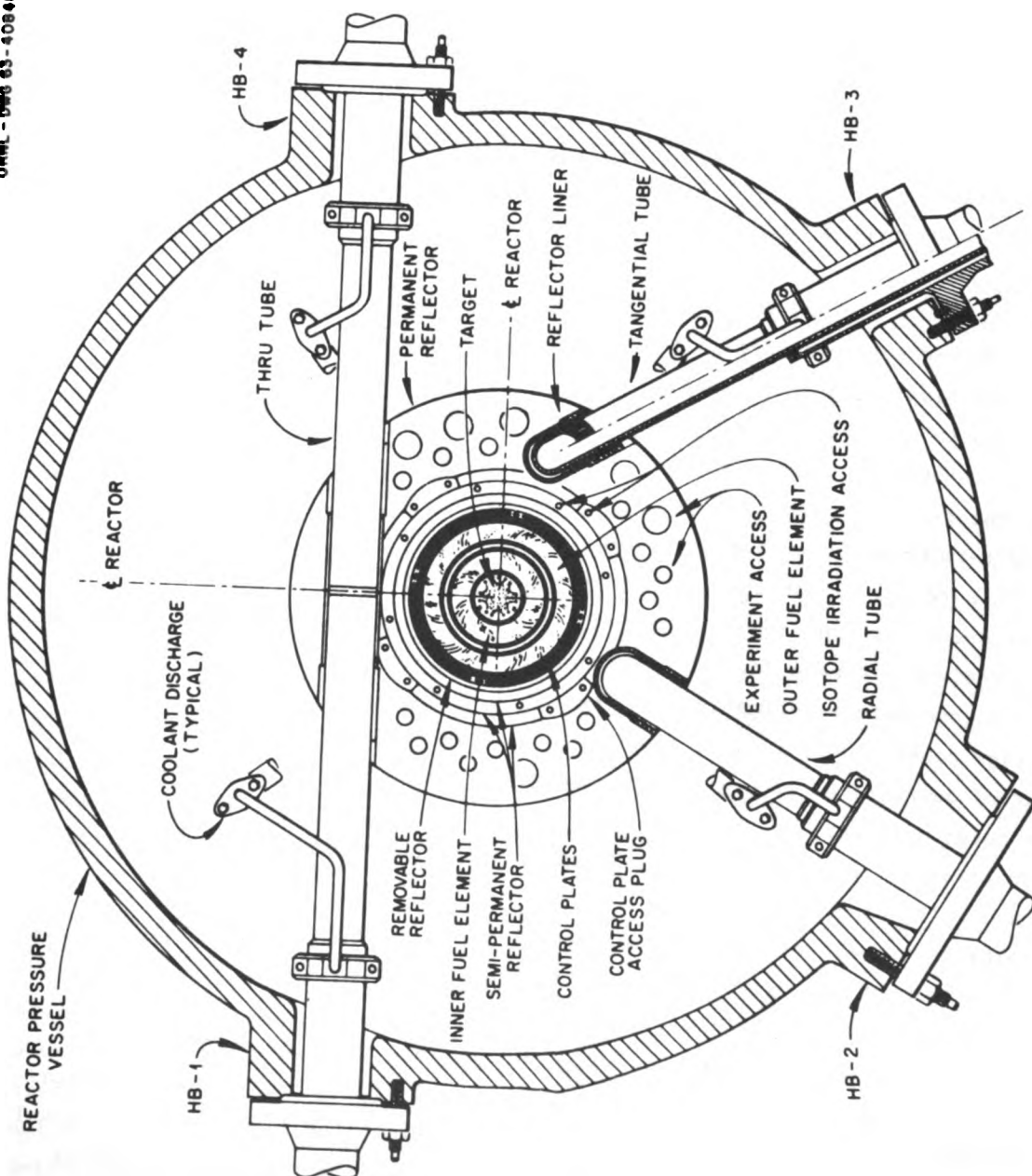


Fig. 6. Horizontal Section Through Pressure Vessel

15 in. (38 cm) from the reactor center line with ends extending outward from the reactor; they are arranged to allow the installation of a single through-tube, should this be desirable in future experimental programs. Design of the beam tube installation was subject to the criterion that they were not to significantly affect the power distribution in the fuel region. In order to meet this criterion it was decided that about three inches of beryllium should separate the fuel region from the beam tubes and the design proceeded on this basis without a significant effort to investigate the optimization of beam tube locations. Early concepts of the design were based on radial orientation of the beam tubes; however, most experiments being considered were of a type using thermal or only slightly epithermal neutrons. Experiments conducted by the Naval Research Laboratory [7] and Brookhaven National Laboratory [8], and calculations at ORNL [9] showed an advantage in terms of reducing the very high energy neutron and gamma rays from the fuel region and therefore three of the tubes were changed to tangential. One radial beam was retained primarily for those experiments in which the higher energy radiation would be of interest.

Each tube is sealed to, and supported by, the reactor pressure vessel by means of a system of clamped and bolted flanged joints. From the flanged connection at the pressure vessel, each tube continues through the reactor pool and pool wall and terminates in a recess located in a large beam port cavity in the shield wall. This recess allows the installation of a shutter and shielding and also provides space for experimental equipment.

Provision is made for the future installation of four engineering facilities to accommodate experiments that require a relatively low neutron flux. These facilities consist of 3-1/4 in. (8.25 cm) ID tubes which enter the pressure vessel and extend downward at an angle of approximately 41° to provide access to the flux at the outer periphery of the beryllium. The upper ends of the tubes terminate at the outer face of the pool wall in the experiment room which is located on the level above the beam tube level.

The permanent reflector is penetrated by 22 vertical holes which extend completely through the beryllium. In addition, 16 similar vertical holes are provided in the removable beryllium.

These vertical holes provide space for various types of static irradiations. In addition, it is anticipated that several of them may be used as locations for future hydraulic facilities which will permit the insertion and removal of samples during reactor operation. When not in use, the vertical facilities will be filled by beryllium plugs sized to provide an adequate cooling annulus between the hole liner and the plug.

10. REACTOR BUILDING AND AUXILIARIES

The main reactor building, associated electrical building, cooling towers,

filter pit, exhaust stacks, and maintenance-office building are shown in Fig. 7. Detailed descriptions of these facilities are given in reference [3].

11. CORE PHYSICS

Because of the desire to achieve unusually high performance in the HFIR, an extensive critical-experiment program was conducted in parallel with the analytical studies. However, since the fuel region consisted of only two fuel elements, and since the radial fuel distribution could not be readily modified, the core was not amenable to parametric-type experimental analysis. It was therefore necessary to rely upon calculations for determining the proper island diameter and fuel region metal-to-water ratio and length-to-diameter ratio, and to some extent the fuel loading and fuel distribution. It was also necessary to rely upon calculations for determining fuel-cycle characteristics such as the power distribution and core lifetime. In these instances, the role of the critical experiment was relegated to that of verifying the adequacy of the calculational methods.

The following parameters were investigated in the critical experiments: power distribution, flux distribution, control rod reactivity characteristics, fuel worth, temperature, void and fuel coefficients of reactivity, plutonium target characteristics, and neutron lifetime.

Most of the analytical studies were made with one-dimensional, multigroup, and two-dimensional, two-group, diffusion theory reactor codes [10,11] because they provided the only practical means for making fuel cycle calculations; good agreement was obtained between analytical and experimental power distributions.

Neutron cross sections used in the above calculations were obtained using a multiregion, thermalization, transport-theory code [12] for the thermal neutrons, and a single-region, multigroup, consistent P_1 theory code [13] for the non-thermal neutrons. The use of the thermalization code made it possible to obtain spectrum-averaged thermal neutron cross sections as a function of radial position in the fuel region.

After the first two critical experiments established the adequacy of the calculational methods, fuel cycle calculations were made to help determine the proper fuel and burnable poison loadings and distributions. The final results indicated that an average fuel cycle time of 14 days could be achieved with an initial U^{235} loading of 9.4 kg, and that the maximum power density and the thermal neutron flux in the target would be essentially constant throughout the cycle. A fuel loading was then fabricated to these specifications and a third set of critical experiments was run to verify the results of the calculations. The element was constructed in such a way that the fuel plate coolant channels could be filled with various solutions in order to uniformly poison the fuel region without changing the condition of the flux trap or reflector regions. This element was also fabricated to correspond, in essential dimensions, to the dimensions speci-

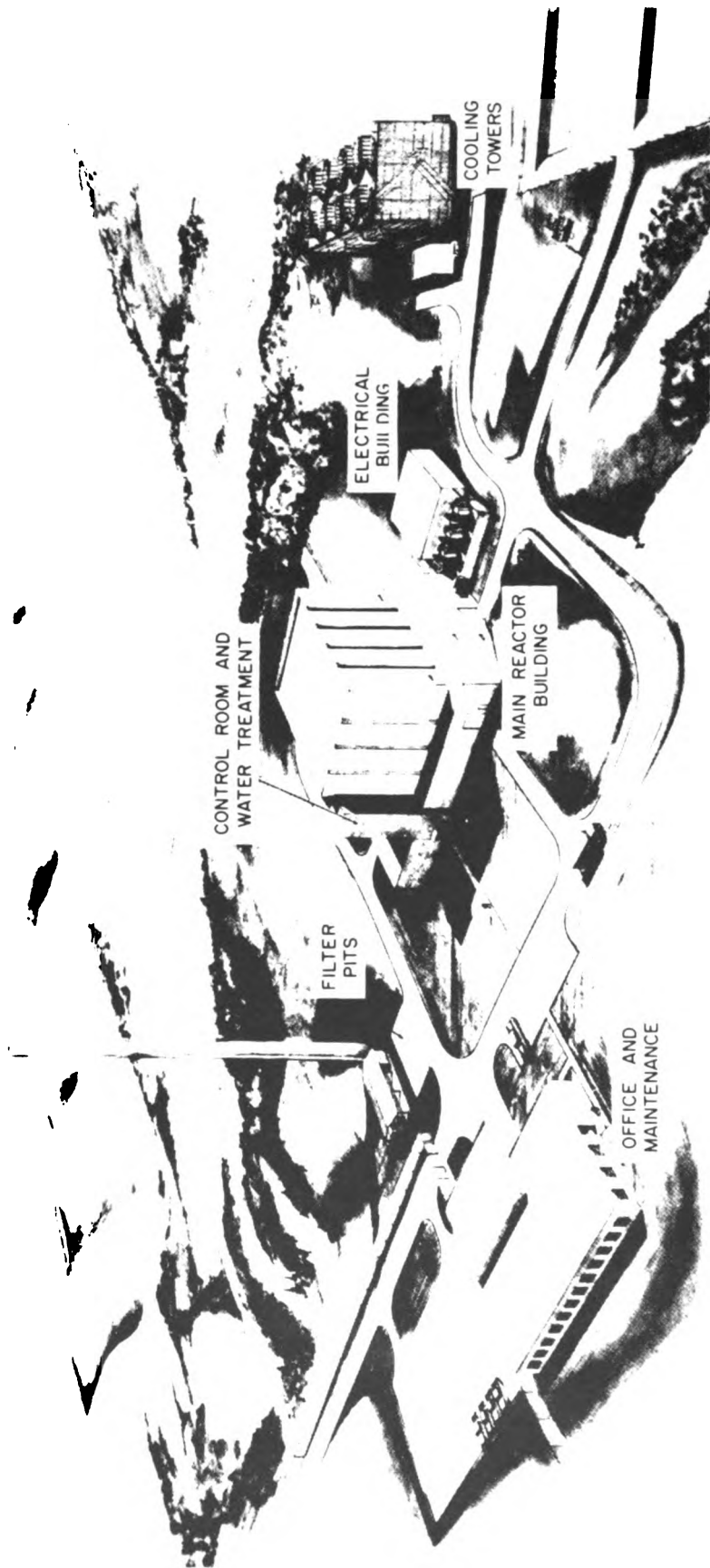


Fig. 7. High Flux Isotope Reactor

fied for the reactor fuel elements in order that it could be used in the reactor for low-power experiments during the startup program. Typical radial neutron flux distributions across the core are shown in Fig. 8.

12. STARTUP AND APPROACH-TO-POWER PROGRAM

The program is divided into two phases and covers operation of the reactor from initial criticality through 100-Mw operation. The first phase covers operation through 20-Mw power level and is termed the startup program. The second phase is termed the approach-to-power program and covers operation from 50 Mw through 100 Mw.

Startup Program

The 20-Mw end point for this phase was selected on the basis that the power level would be sufficiently high to bring all normal instrumentation into operation while still being sufficiently low that the heat transfer requirements were essentially negligible in terms of the design.

On August 25, 1965, the HFIR achieved criticality for the first time, and on January 29, 1966, the power level was increased to 20 Mw. Operation at or near 20 Mw was continued until May 31, 1966, at which time the power level was increased to 50 Mw.

During the period from August 25, 1965, to January 29, 1966, an extensive nuclear "zero power" experimental program was conducted for the purpose of investigating the nuclear characteristics of the actual HFIR facility, for compiling generally useful data associated with the routine operation of the reactor, and for operator training.

The types of experiments conducted for the purpose of investigating the nuclear characteristics are as follows:

- (1) Reactivity shutdown margins.
 - (2) Control rod reactivity differential worth.
 - (3) Fuel, fuel plate, and void reactivity coefficients for the fuel regions.
 - (4) Isothermal temperature reactivity coefficients.
 - (5) Reactivity worths of various targets and voids in the flux trap.
 - (6) Reactivity worth of water in the beam tubes.
 - (7) Reactivities associated with replacement of beryllium reflector vertical experimental facility beryllium plugs with water and voids.
 - (8) Effect of flow and pressure on reactivity and reactivity stability.
 - (9) Reactivity worth of B¹⁰ stainless steel strips in the fuel elements (secondary shutdown).
 - (10) Power distributions.
 - (11) Response of safety system to neutron-generated signals.
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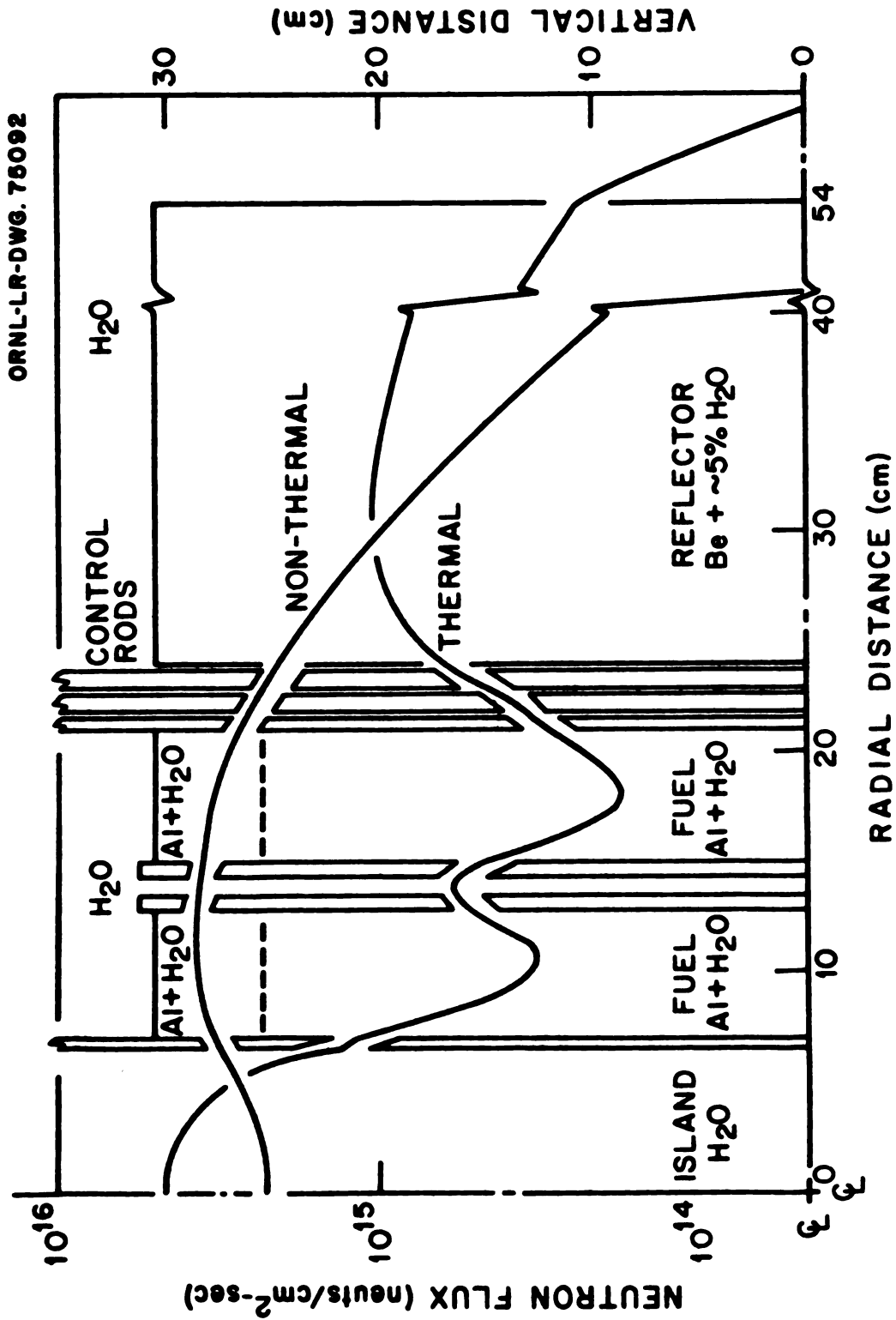


Fig. 8. Core Schematic and Flux Distribution
(Clean Core at 100 Mw)

In the zero power experiments several combinations of fuel elements, control rods, and targets were used. The fuel elements included the critical experiment element (9.4 kg U^{235} , 2.12 g B^{10}), which is referred to as the HFIRCE-3 element, the ORNL production element (9.4 kg U^{235} , 3.60 g B^{10}), and the first element fabricated by Metals and Controls, Inc., (9.4 kg U^{235} , 2.80 g B^{10}). The only significant difference between these elements is the loading of the B^{10} burnable poison. Subsequent elements are presently specified to contain 2.80 g B^{10} .

Two sets of control rods were used: the HFIRCE-3 rods and the first production rods. With the exception of a few hydraulic relief holes these two sets of rods are identical.

The different targets used in the flux trap are listed in Table III. To avoid the possibility of voids entering the flux trap during the early experiments, two different plastic containers were used to displace the water in the flux trap. One contained the HFIRCE-3 simulated 300 g Pu^{242} target* plus the corresponding optimum void space; the other was provided with the optimum void space that corresponds to no target in the flux trap. Both of these plastic containers could be voided or filled with water.

Table III
Targets Used in Flux Trap

Nomenclature	Description
PI + W	Plastic container filled with water
PI + V	Plastic container with optimum void
PT + W	Plastic container with HFIRCE-3 target and filled with water
PT + V	Plastic container with HFIRCE-3 target and optimum void
DFTT	Dummy flow test target (all aluminum)
U^{235} Rods	Standard target using U^{235} , U^{238} , Ta to simulate maximum heat generation rate
Pu^{242} Rods	Standard target containing 248 g Pu^{242} , 2.5 g Pu^{239} , 2.5 g Pu^{241}

In view of the fact that much time and effort was spent on HFIR critical experiments at the critical facility, there is perhaps some question as to why so many experiments were proposed for the HFIRCE-3 fuel element and control rods in the HFIR facility. There are two main reasons. One is that near the completion

*The HFIRCE-3 target contained U^{235} , U^{238} , Ag to simulate a maximum reactivity 300 g Pu^{242} target.

of the HFIRCE-3 experiments a change was made in the design of the coolant channels in the control region of the HFIR. The effect of this change was to decrease the worth of the control rods; thus it seemed prudent to rerun many of the rod

calibration experiments in the HFIR facility.

The other reason stems from the fact that the HFIRCE-3 experiments were terminated before all of the desired power distribution data were obtained. The data of particular interest is that which can be used to predict the power distribution variations in the vicinity of the control rods (window peaking effects) for various control rod positions. Although this additional information is not necessary for evaluating the present fuel element design, it will be needed for extrapolation to more heavily loaded elements should such changes be desirable in the future.

There are, of course, other reasons for conducting the HFIR experiments such as operator training and compilation of generally useful data associated with the routine operation of the reactor. Furthermore, there are "reactivity" experiments associated with coolant flow and pressurization that could not be performed in the HFIRCE-3 facility.

Differential worth of the control rods was determined for different rod positions by maintaining criticality with the rods (asymmetrical rods) and also by supplementing the control with boron in the confined moderator (~symmetrical rods). The former type of rod calibration was performed at the HFIR facility before any of the boron experiments were conducted. Therefore, these experiments provided data with which to compare the reactive state of the HFIRCE-3 and HFIR facilities. A summary of these data is presented in Fig. 9. As indicated, the agreement between the two facilities is very good. A comparison of the dashed curves, representing the HFIRCE-3 data, with the solid curves presumably shows the effect of the different coolant channels in the control regions of the two facilities. As was expected, this difference tends to reduce the worth of the safety rods in the HFIR facility. The differences observed in the differential worth curves are probably within the degree of accuracy associated with the experiments; however, the rod position data should be accurate enough to indicate a real difference.

Power distributions were determined in essentially the same way as was done during the HFIRCE-3 experiments, using the same removable fuel plates which contained "punchings" used as fission foils. These punchings were made at selected locations and had a fuel content typical of the fuel loading at that location. The foil (punching) weights (weight of U^{235}) were determined using a scintillation counter. Relative power distributions were obtained by comparing each foil's total gamma activity to the time-interpolated activity of a normalizing foil that had been irradiated at the same time and was counted periodically during the counting of the other foils.

The results of the power distribution measurements confirmed the work done at the critical facility and provided the additional information needed for studies of possible future loadings. A few general comments regarding azimuthal

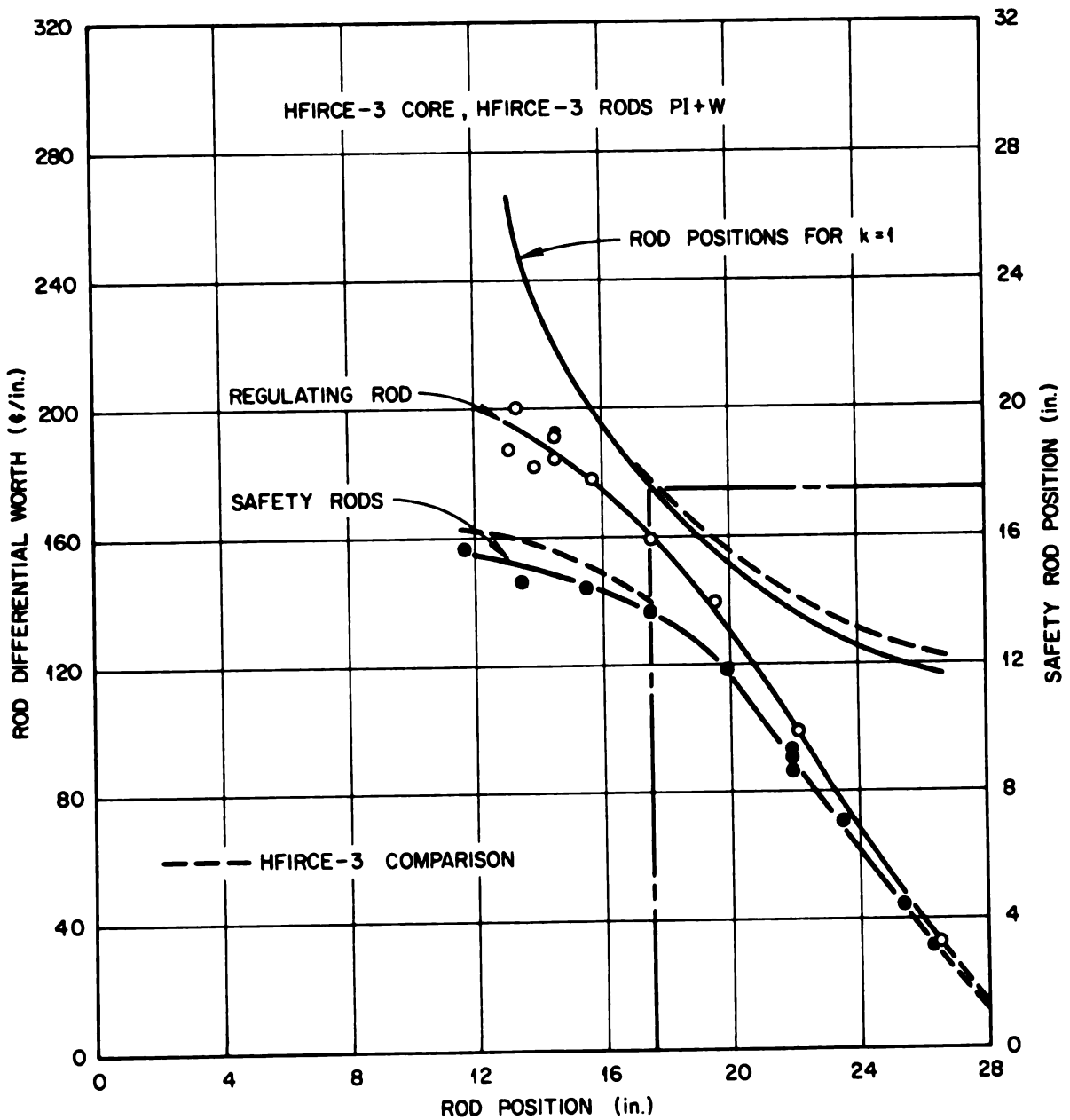


Fig. 9. HFIR Differential Rod Worth Data

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and longitudinal symmetry may be of interest. There is essentially no effect of beam holes on power distribution, but the window between control rod poison quadrants and the greater water concentration in that area increases the local power density by as much as 25% and by no less than 5%. The latter value exists with the rods fully withdrawn, in which case the peaking presumably results from the greater water concentration associated with the stationary divider strip between the safety rods. The largest peaking occurs with the rods in their innermost critical position (clean core plus maximum fission target). These effects will have to be considered when evaluating the overall window peaking problem for future cores. However, the present core design has sufficient margin in the area of interest to accommodate these local peaks since they are no different than observed in previous experiments.

In the longitudinal direction there is a tendency for the power density to be greater at the coolant inlet end than at the outlet because of the slight longitudinal asymmetry associated with the radial displacement of the shim-safety rods relative to the shim-regulating rod. The effect is more pronounced for the clean core condition than for the fully withdrawn-rod condition because as the rods are withdrawn the "reflector" asymmetry decreases. A preliminary analysis of the data indicates that for the clean core, symmetrical rod condition the power density at the outlet end of the core is ~17% less than at the inlet end for the inner element and ~28% less for the outer element. When the rods are fully withdrawn, the corresponding values are ~3% and ~11%. This agrees reasonably well with previous data and even after making a correction for inaccuracies indicates a possibly significant improvement in heat removal in the early part of a fuel cycle.

Fuel and void reactivity coefficients were obtained for the inner and outer fuel elements by replacing regular fuel plates with water and by replacing non-fuel-bearing aluminum fuel plates with water. These plates occupied the regular removable fuel plate slots; there were six nearly equally spaced inner and outer element plates involved in both tests. The reactivity worth of replacing these plates with water was obtained by withdrawing the appropriate number of plates with the reactor initially critical and determining the positive stable period. Table IV summarizes the reactivity worth of several changes in target arrangements in the reactor and the fuel, fuel plate, and void reactivity coefficients of the inner and outer fuel elements.

The reactivity changes resulting from flooding of the beam tubes and replacement of the reflector beryllium plugs with water and voids, as shown in Table V, are negligible.

Table IV

Summary of Reactivity Worths and Coefficients

$(PT + W) - (PI + W)^a$	$+ 0.0070 \Delta k/k$
$(PT + V) - (PT + W)^a$	$+ 0.015 \Delta k/k$
$(PI + V) - (PI + W)^a$	$+ 0.032 \Delta k/k$
Fuel Coefficient^b	
Inner Element	$+ 3.72 \times 10^{-5} \frac{\Delta k/k}{g U^{235}}, + 0.0966 \frac{\Delta k/k}{\Delta m/m}$
Outer Element	$+ 1.09 \times 10^{-5} \frac{\Delta k/k}{g U^{235}}, + 0.0744 \frac{\Delta k/k}{\Delta m/m}$
Total Average	$+ 0.171 \frac{\Delta k/k}{\Delta m/m}$
Fuel Plate Coefficient^c	
Inner Element	$- 0.0458 \frac{\Delta k/k}{\Delta v/v}, - 8.72 \times 10^{-5} \frac{\Delta k/k}{in.^3}$
Outer Element	$- 0.113 \frac{\Delta k/k}{\Delta v/v}, - 11.6 \times 10^{-5} \frac{\Delta k/k}{in.^3}$
Fuel Region A1 Coefficient^d	
Inner Element	$- 0.104 \frac{\Delta k/k}{\Delta v/v}, - 16.7 \times 10^{-5} \frac{\Delta k/k}{in.^3}$
Outer Element	$- 0.188 \frac{\Delta k/k}{\Delta v/v}, - 15.9 \times 10^{-5} \frac{\Delta k/k}{in.^3}$
Fuel Region Void Coefficient^e	
Inner Element	$- 0.080 \frac{\Delta k/k}{\Delta v/v}$
Outer Element	$- 0.170 \frac{\Delta k/k}{\Delta v/v}$

(a) Refer to Table III for nomenclature.

(b) m refers to weight of fuel in specified region.

(c) v refers to active volume of fuel plate.

(d) v refers to the entire volume of fuel plate.

(e) v refers to the volume of water.

Table V
Reactivity Effects Due to Changing Experimental Facilities

Core Condition	Symmetrical Critical Position	Reactivity Change ^a (¢)
Standard core - all Be plugs in	16.311	-
Standard core - all Be plugs out	16.332	- 4.5
Standard core - six air filled plugs	16.328	- 4.0
Standard core - all beam holes empty	16.310	-
Standard core - HB-1 filled	16.306	+ 1.05
Standard core - HB-1, 2, filled	16.305	+ 1.68
Standard core - HB-1, 2, 3, filled	16.302	+ 2.40
Standard core - HB-1, 2, 3, 4, filled	16.302	+ 2.10

^a Rhoette reading

The only result from all these experiments that deviates significantly from what was expected is that associated with the isothermal temperature coefficient. In the HFIR critical experiments the isothermal temperature coefficient was found to be slightly negative over the expected temperature range (68-160°F), whereas that measured in the HFIR facility was slightly positive up to about 120°F, with a maximum increase in reactivity from room temperature of 0.07% $\Delta k/k$. This constitutes no particular operating problem since the fuel region temperature coefficient is strongly negative. The difference in the two measured isothermal coefficients is attributed to a difference in the control rod drive mechanisms, a difference in the water content of the control region, which has a slightly positive coefficient, and a difference in the fuel elements (an 8-kg core with uniform burnable poison distribution was used in the original measurement).

In addition to the measurements described above, several sets of flux measurements were made in the flux trap and beryllium irradiation facilities. These measurements were made for two purposes: first, to measure the thermal, resonance and fast flux in the various facilities; and second, to obtain information on the neutron energy and intensity distribution to allow comparison with calculations set up for use by operations personnel and experimenters. The reduction of data from these irradiations has not yet been completed; however, preliminary results extrapolate to a perturbed thermal neutron flux of 2.6×10^{15} at 100 Mw at the center of the flux trap with the trap loaded with Al dummy rods. This flux is based on the irradiation of cobalt and gold monitors.

A joint Argonne National Laboratory and Oak Ridge National Laboratory experiment on the quality of neutron beams from the HFIR beam tubes was conducted. The

leakage radiations from the HFIR are not expected to be greatly different than those from the Argonne Advanced Research Reactor (AARR) which is being designed, and therefore information regarding the beam quality is of interest in the AARR program and for use in planning experiments.

While the data from these experiments have not been completely analyzed, the following preliminary results are presented due to the interest in the design of beam tubes which are better "tailored" to the needs of the experimenters.

Experiments were performed using two of the horizontal beam tubes, HB-2 (radial) and HB-3 (tangential). Neutron-spectrum measurements for both radial and tangential tubes should reveal the extent of the improvement (greater slow-neutron to fast-neutron ratios and fewer gamma rays) in beam quality as the line from beam origin to experiment ceases to intercept the fission source.

Ionization chamber measurements indicate a gross gamma ray intensity for HB-2 (radial) about nine times that for HB-3. For the full beam tube (no added collimation), the extrapolated intensity for HB-2 at 100 Mw is about 2×10^5 r/hr. Comparable relative intensities were obtained with a scintillation spectrometer; however, the relatively strong Be-capture contribution observed in the HB-3 beam was the most interesting feature.

Both foil-activation and U^{235} fission counting indicate thermal neutron currents only about 5% higher for HB-2 than for HB-3. For an essentially uncollimated beam, the thermal neutron current at a distance of six feet from the shutter exit, on the beam tube axis, extrapolates to about 7×10^9 for 100-Mw operation. Comparison of this with the current observed at a typical ORR (30-Mw) beam tube experiment and making appropriate allowance for the degree of collimation indicates a factor of 4-10 advantage for the HFIR (100 Mw) in terms of benefit for experiments of the types presently planned. This possible advantage does not take into account the improvement in signal to background ratio anticipated for the tangential beam tubes.

Measurements taken with a boron-absorption spectrometer indicate roughly equal currents for the two beam tubes in the energy range 1.0 to 100 ev. Proton-recoil spectrometer data shows roughly comparable intensities in the kev region with quite strong structures due to Al resonances, presumably due to the several Al diaphragms in the Al beam tube.

Measurements made using fission counting techniques show comparable intensities for U^{233} , U^{235} , and Pu^{239} and factors of 1.51 for Np^{237} (0.4 Mev), 1.59 for U^{236} (0.7 Mev), and 1.56 for U^{238} (1.3 Mev) for the intensity of HB-2 (radial) over that for HB-3 (tangential). Threshold detectors for the range of 2.9 to 8.1 Mev yielded factors of 5-10.

Measurements were also made by stopping each beam in a large water tank and making thermal neutron flux measurements in the forward direction. The re-

sults indicate about a factor of 10 difference in gross Mev-neutron intensity.

In general the measurements confirm the predicted advantage of the tangential beam tubes over radial beam tubes. It appears that the tangential beam gamma intensity would be somewhat lower in a D_2O reactor where the H- and Be-capture gamma rays should not be present. Also it would be anticipated that fewer neutrons in the Mev range would be present in the beam from a D_2O system due to the more nearly elastic scattering of the Be in contrast to that of deuterium.

Operation at power levels up to 20 Mw was continued during the period from January 29, 1966, through May 30, 1966. Many tests were conducted to obtain operational information and to verify proper behavior during simulated electrical power outages, etc.; however, much of the time was devoted to straightforward operation while awaiting final safeguards approval, from the USAEC, for operation at power levels above 20 Mw. The accumulated power on the first fuel loading (and reactor) was approximately 1300 megawatt-days at the end of this period.

Approach-to-Power Program

The approach-to-power phase of the program consists of operation of the reactor for a full fuel cycle at each of several successively higher power levels. As noted previously, the main areas of concern are those influencing fuel plate temperature and influenced by plate temperature, i.e., aluminum-oxide film buildup and radiation damage effects.

The first step in power level was selected on the basis that no significant effects should be observed. This step was originally specified to be 50 Mw for a fuel cycle; however, due to the long period of operation at 20 Mw it was decided to simply complete the fuel depletion at 50 Mw and then proceed to 75 Mw operation with the second loading.

On May 31, 1966, the power level was increased to 50 Mw and operation was continued, still using the first fuel loading and still with no experiments installed. Operation continued at 50 Mw until June 19, 1966, at which time the control rods were fully withdrawn and shutdown occurred due to fuel depletion. The total integrated power at shutdown was about 2230 Mwd; this corresponds to an average U^{235} burnup of about 30%. This is somewhat in excess of the 1800-1900 Mwd estimated for the clean conditions which were present. Operation during the 50-Mw run was uneventful and no shutdown occurred other than one scheduled test of the safety rods shortly after reaching 50 Mw. No effects were observed which indicated any significant change in fuel plate dimensions due to oxide buildup or radiation damage.

The reactor was reloaded and following several experiments to check on various operational conditions, the reactor was taken to 75 Mw on June 30, 1966. With the exception of a power reduction to about 30 Mw for a few minutes when one

of the main pumps was tripped off the line during a severe thunderstorm, the reactor operated steadily at 75 Mw until July 31, 1966, at which time fuel depletion caused shutdown. The accumulated power at shutdown was 2310 Mwd. Evidence of some change in the fuel plates occurred during this run. The fuel element pressure drop increased about 5% and the flow decreased about 5% during the cycle.

The next operating step in power level is scheduled for 90 Mw and if all continues in a satisfactory manner, 100 Mw should be reached in September of 1966.

13. BEAM TUBE EXPERIMENTS

Three of the horizontal beam tubes have been assigned to the Solid State Division and the fourth is assigned to the Chemistry Division. Two of the Solid State Division experiments and the Chemistry Division experiment will consist of triple-axis neutron crystal spectrometers which will be entirely automatic and which will be controlled by computers (PDP-8). One of the Solid State units will be used primarily to obtain added sensitivity in the investigation of certain magnetic properties of materials. The other one will be used primarily for experiments involving the inelastic scattering of neutrons to obtain information on atomic and molecular energy levels and on the dynamical properties of crystal lattices. The third neutron diffraction setup (Chemistry Division) will be used for crystal structure analysis with emphasis on biological molecules. The fourth beam tube experiment (Solid State Division) is currently in the proposal stage and present plans are to install a neutron spectrometer consisting of a neutron chopper with associated time-of-flight measuring equipment for use in those investigations of solid state interest that can not be readily investigated with a triple-axis crystal spectrometer.

It is anticipated that the three triple-axis spectrometers will be in service by the end of this year. The time-of-flight spectrometer, if approved, should be in service by the end of 1967.

14. RADIOISOTOPE PROGRAM

In addition to the irradiation of Pu^{242} in the flux trap to produce trans-uranium isotopes, it is planned to use the HFIR for the preparation of other high specific activity isotopes. The preparation of these isotopes will provide additional tools for solving medical and physical research problems. Although the vertical irradiation holes in the beryllium may be used, most of the irradiations are planned to be done in a hydraulic rabbit tube facility located in the flux trap on the vertical centerline of the core. The hydraulic system is designed and is presently being tested out of the reactor. Installation will be made following completion of the approach-to-power program for the reactor.

15. COSTS

Table VI lists the approximate capital expenditures incurred in the design

and construction of the HFIR. Costs for equipment associated with beam tube experiments, etc., are not included in this breakdown.

Table VI

Design and Construction Costs of HFIR

Engineering, design, and inspection	\$ 3,175,000
Reactor (core, reactor vessel, cooling system, etc.)	4,150,000
Instrumentation	1,300,000
Building (site preparation, structures, electrical)	<u>6,100,000</u>
	\$14,725,000

The development costs incurred in the HFIR program total about $\$6.7 \times 10^6$, of which about $\$2.5 \times 10^6$ was expended on fuel element and control plate development.

Operational costs totaling about $\$6.8 \times 10^6$ were expended up to July 1, 1966. Of this amount about $\$3.6 \times 10^6$ was expended in preoperational and low power testing, training of the operating staff and miscellaneous technical support. About $\$2.3 \times 10^6$ was spent on fuel element procurement and the remainder for procurement of spare parts and miscellaneous materials.

Table VII lists the estimated annual operating cost including the ORNL overhead and the cost of fabricating 24 fuel loadings, but not including the cost of the uranium burned or the cost of reprocessing the depleted fuel elements to recover the unburned uranium. Costs, other than incidental services by operations personnel, associated with experimental programs are not included in this estimate of operating costs.

It is estimated that each of the beam tube experimental setups will cost about \$215,000 for equipment and installation, with an estimated annual operating cost for each program of about \$100,000. The estimated cost of fabricating and installing the central hydraulic rabbit tube and loading station is \$50,000.

16. POTENTIAL OF HFIR-TYPE REACTORS FOR HIGHER FLUXES

Several preliminary studies have been undertaken to assess the possibility of achieving fluxes significantly higher than 10^{16} neutrons/cm² sec in the flux trap of a reactor similar to the HFIR. One study (unpublished) by R. D. Cheverton of the HFIR Project indicates that a five-fold increase in the power density to ~10 Mw per liter is probably ultimately possible. With a modest increase in core size and the use of D₂O as a coolant, it appears that an unperturbed thermal neutron flux in the island of about 3×10^{16} would be achieved at a power level of about 750 Mw in this "ultimate" HFIR-type reactor. In order to achieve this sort of performance significant improvements would have to be made in fuel element

Table VII
Estimated Annual Operating Expense of HFIR^a

Reactor Operating Personnel	\$ 425,000
General Operating Costs (Utilities, Liquid Waste Handling, Health Physics, etc.)	475,000
Instrumentation and Controls Division	380,000
Craft and Engineering Support	
Mechanical Craft and Engineering Support	450,000
Analyses and Technical Support	140,000
Miscellaneous Material Costs	120,000
Spare Components	100,000
Control Plate Fabrication	210,000
Fuel Element Procurement ^b	<u>1,500,000</u>
Total	<u>\$3,800,000</u>

^aIncludes ORNL overhead, etc.

^bThe unit cost used here is \$62,000 per loading. This is an optimistic figure as present experience indicates a cost of about \$95,000 per loading for the first 33 loadings and about \$75,000 for the next 24 loadings.

technology; however, the necessary improvements do not appear to require "inventions" but extensions of techniques currently under investigation. It is highly questionable whether such a system could be considered feasible at this time from the standpoint of fuel cycle and pumping power considerations. However, the study certainly serves to place a reasonable upper limit on the probable maximum performance of this type of reactor at the present time and probably for some time to come.

In addition to consideration of this "ultimate" HFIR-type reactor, Mr. Cheverton also considered the question of whether or not one could build an HFIR-type reactor to produce an unperturbed thermal neutron flux of 1×10^{16} , using only a modest extension of current technology. It was concluded that such a system could be built and that only a modest development effort would be required.

17. ACKNOWLEDGMENT

It is manifestly impractical to give recognition in this paper to all of the ORNL personnel who have participated in the development, design, construction,

and initial operation of the HFIR facility; however, the following are noteworthy among those who have made major contributions: project directors, C. E. Winters (1960-1961), A. L. Boch (1961 to present); physics and general reactor engineering, R. D. Cheverton; heat transfer and hydraulics, N. Hilvety, H. A. McLain, and T. G. Chapman; shielding and general reactor engineering, H. C. Claiborne; reactor design engineering, J. R. McWherter, J. H. Westsik, R. E. Hoskins, W. G. Cobb, and R. E. Schappel; fuel element and control plate development and fabrication, G. M. Adamson and C. F. Leitten; reactor control and instrumentation, L. C. Oakes; construction coordination, J. W. Hill, Jr. and H. Grimac; reactor component fabrication coordinator, R. M. Hill, Jr.; reactor operations supervisor, R. V. McCord.

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THE ARGONNE ADVANCED RESEARCH REACTOR (AARR)*

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ABSTRACT

The Argonne Advanced Research Reactor (AARR) is being designed as a multipurpose research facility for experiments requiring high-intensity fluxes of thermal neutrons. It will be located at the DuPage County, Illinois, site of Argonne National Laboratory so as to be close to facilities at Argonne and to be readily accessible to scientists at Argonne and from the community of midwest universities. Initial power operation is scheduled for 1970.

The entire reactor and the experimental facilities will be housed within a reinforced-concrete steel-lined Reactor Containment Building. The AARR plant will include also a Laboratory and Office Building, and other buildings and supporting facilities.

The experimental facilities will include: a central vertical thermal column (ITC), containing a total of 9 static and rabbit vertical tubes for target irradiations; a total of 8 horizontal tangent tubes (6 blind beam tubes, and 2 through tubes), which penetrate the outer radial beryllium reflector to the proximity of the reactor core; and a total of 19 vertical irradiation facilities in the beryllium reflector.

The design fission power of the reactor is 100 MWt. At this power level, unperturbed peak thermal neutron fluxes of 4×10^{15} in the ITC, and 1×10^{15} at the tips of the beam tubes, are expected. The primary purpose of the ITC is to support research in nuclear chemistry, and especially for generation of research quantities of very heavy (transuranic) isotopes. The horizontal tubes in the beryllium will be used for: fundamental research in nuclear physics and particle physics; nuclear spectroscopy; and neutron diffraction and inelastic scattering studies of matter. The irradiation facilities in the beryllium will be used for chemical and metallurgical research activities.

* Work performed under the auspices of the U. S. Atomic Energy Commission

The reactor is a pressurized-water (H_2O) system of the heterogeneous, flux-trap type. The annular fuel zone is made up of assemblies of SS-clad flat fuel plates, with a SS- UO_2 cermet as the matrix material. The UO_2 is highly enriched in U^{235} . The fuel loading is graded, plate by plate, at the inner and outer boundaries of the reactor core, to reduce the power peaks which result from inleakage of slow neutrons from the moderator regions outside the core. The fuel plates have an active height of 46 cm (18 in.), and the volume of the reactor core is approximately 78 liters. The reactivity is controlled by a combination of a burnable neutron poison, five safety and one power-regulation control blade at the outer radial perimeter of the fuel zone, and six shim control blades in the interior of the fuel zone.

Light water (H_2O) serves as the coolant/moderator in the reactor fuel zone and in the Internal Thermal Column. In the reactor core, the volume ratio of water/non-water is slightly less than unity, primarily to provide a large heat-transfer surface for operation at high power densities. The core is heavily loaded (~ 60 kg U^{235}) to attain a long core life, estimated to be 90 24-hr days at 100 MWt. The neutron flux spectrum in the bulk of the fuel zone may be characterized as "low-intermediate", with a median neutron energy of ~ 20 eV for fissioning in the fresh reactor. Thus, even at these high power densities, shutdown xenon can be overridden during most of the core life.

The construction budget for the AARR is \$25 million, including the plant, the Architect-Engineer design costs, contingency funds, and \$1 million for experimenters' equipment. The annual operating cost, for 24-hour/day usage at 100 MWt, is estimated to be $\$4\frac{1}{2}$ million. Of this, $\$2\frac{1}{2}$ million per year is for replacement of reactor fuel, control rods, and portions of the beryllium reflector, and the balance is for all other operating costs.

1. Introduction and Design Summary*

The Argonne Advanced Research Reactor (AARR) will be constructed at the DuPage County, Illinois, site of Argonne National Laboratory. Groundbreaking is scheduled for 1967. Initial power operation, scheduled for 1970, will mark a culmination of many years of effort by the Laboratory to obtain a multi-purpose research facility providing high-intensity sources of neutrons. The AARR will provide scientists at Argonne and from the community of Midwest universities with an effective tool for advanced experimental research in a number of scientific disciplines, including nuclear physics, particle physics, nuclear chemistry, solid state science, and metallurgy.

The AARR facility will be comprised of: a cylindrical, reinforced-concrete, steel-lined Reactor Containment Building (RCB), which will house the reactor pressure vessel, the entire primary coolant system, and the experimental

facilities; a Laboratory and Office Building; and other buildings and supporting facilities. The AARR site is just northwest of Argonne's CP-5 reactor. An artist's sketch of the AARR is shown in Figure 1. Figure 2 is a "cutaway" view of the RCB and the Active Material Handling Building.

Reactor facilities for experiments will include:

(a) an internal thermal column (ITC), containing rabbit tubes and tubes for irradiation of fixed targets in thermal neutron fluxes in the 10^{15} - 10^{16} range; at 100 MWt, the unperturbed peak flux in the ITC is computed to be approximately 4×10^{15} n-cm/cm³-sec;

* The information supplied here is based upon the current, or reference design. Title-I design of the plant is being completed now, and the detailed design work of Title II will begin shortly. Because the reactor design is in progress, there are no good current references for details of the engineering design. Therefore, a brief summary is included in this paper.

1. REACTOR CONTAINMENT BUILDING
2. LABORATORY AND OFFICE BUILDING
3. ACTIVE MATERIAL HANDLING BUILDING
4. ELECTRICAL EQUIPMENT BUILDING
5. COOLING
6. RETENTION PONDS
7. 250 ft EXHAUST STACK
8. ARGONNE RESEARCH REACTOR (CP-5)

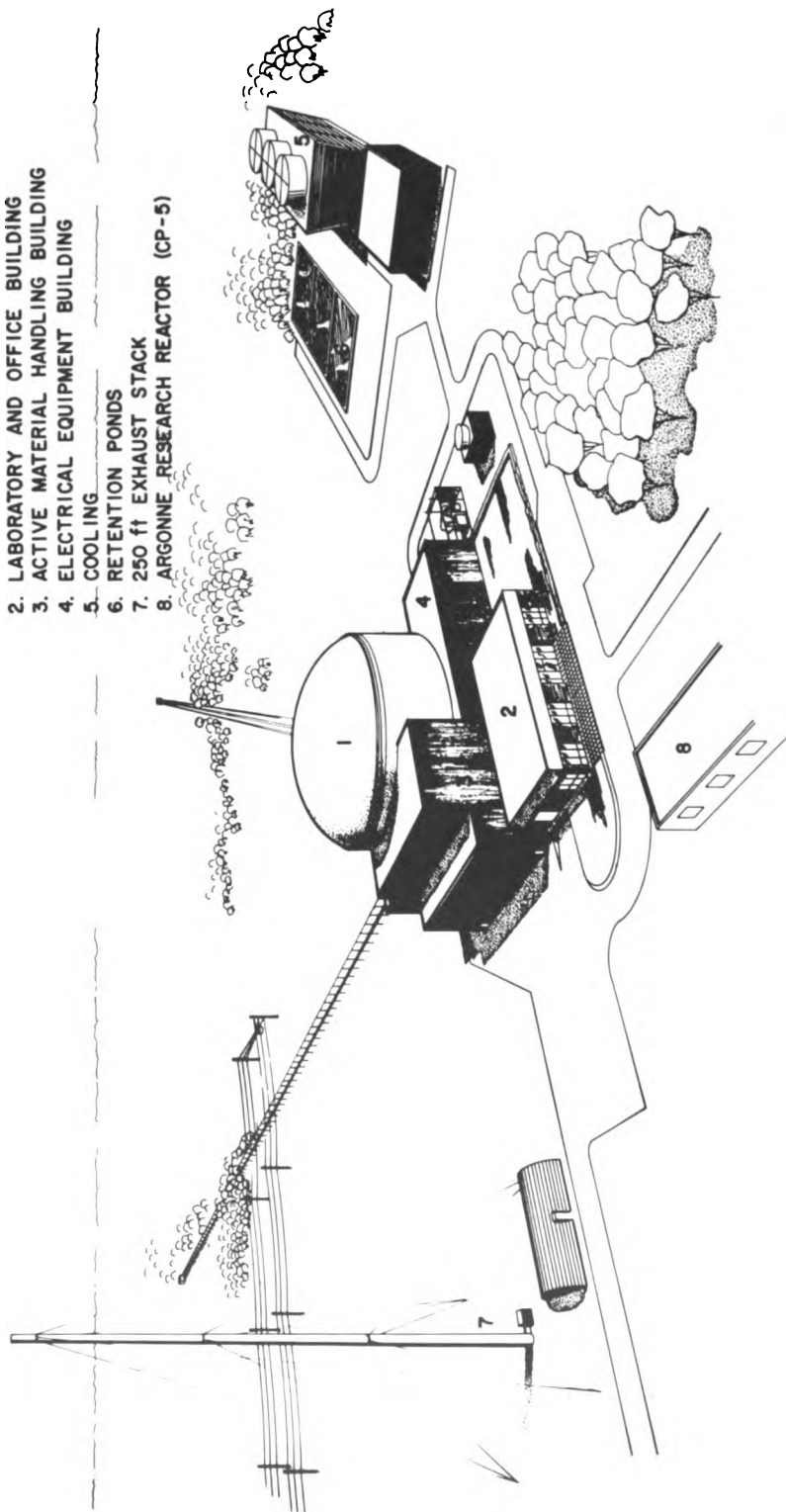


FIG. 1 ARTIST'S CONCEPT OF ARGONNE ADVANCED RESEARCH REACTOR

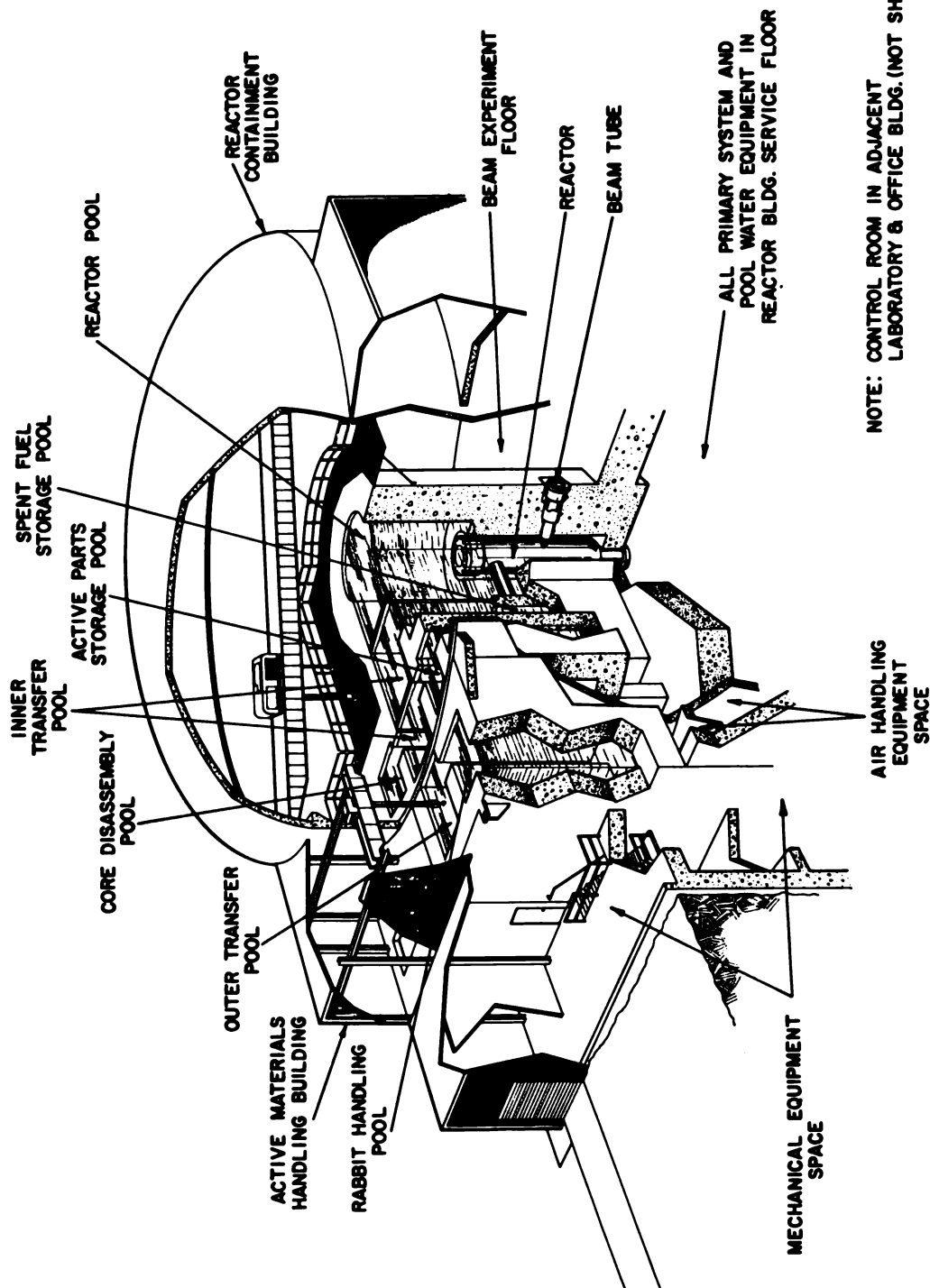


FIG. 2 CUTAWAY OF AARR REACTOR CONTAINMENT AND ACTIVE MATERIAL HANDLING BUILDINGS

(b) a number of horizontal tangent beam tubes and through tubes which penetrate the external beryllium reflector to the proximity of the fuel-zone annulus; it is intended that each "blind" beam tube will provide two separate beams of neutrons at the working face of the lateral biological shield; at 100 MWt, the unperturbed peak thermal neutron flux in the beryllium is estimated to be 1×10^{15} , and the blind beam tubes will terminate in this region of peak flux;

and

(c) a number of vertical irradiation tubes, some of which will be rabbit tubes, in the beryllium; the thermal neutron flux levels in this zone will be much in excess of 10^{14} .

The internal flux-trap zone (ITC) will be used primarily in support of research in the field of nuclear chemistry, and especially for production of very heavy (transuranic) isotopes. The high-intensity fields of thermal neutrons in the horizontal tubes will be applied to: fundamental research in nuclear physics and particle physics; a general program of nuclear spectroscopy, investigating gamma rays associated with capture of neutrons by matter; and studies of crystalline and liquid matter by techniques of neutron diffraction and neutron inelastic scattering. The vertical irradiation facilities in the external beryllium reflector will be used for irradiations in high-intensity fields of thermal neutrons, supporting research in disciplines of chemistry and metallurgy.

The reactor is a pressurized-water reactor of the heterogeneous, flux-trap type, with an annular fuel zone, a central or "internal" thermal column (ITC) and an external radial beryllium reflector. Light water (H_2O) is the coolant throughout the reactor; it serves a dual purpose as the neutron moderator in the fuel zone and in the ITC. The annular fuel zone is made up of 36 rhomboidal assemblies of SS-clad flat fuel plates, with a fuel-plate matrix of SS and UO_2 highly enriched in U^{235} . The fuel loading is graded, at the inner and outer hexagonal boundaries of the reactor core, to reduce the power peaks which result from inleakage of thermalized neutrons from the moderator regions outside the core. The active height of the fuel plates is 46 cm (18 in.), and the volume of the reactor core is approximately 78 liters, including the volume of the in-core control-rod channels. The reference power level is 100 MWt.

The reactivity of the system is controlled by a combination of a fixed burnable neutron poison and a set of 12 flat control blades. Six of the 12 blades are used for reactivity shimming; they are in the interior of the reactor core. The other 6 blades border the core at its outer perimeter; they are safety blades, except for one blade which serves the dual purpose of a power-

regulation blade for automatic control of reactor power. The primary use of the peripheral control blades is as safety blades, so as to maximize the thermal neutron fluxes in the experimental facilities in the beryllium reflector.

The external radial reflector includes a 12-in.-thickness of beryllium, with coolant passages, vertical holes for irradiation tubes, and horizontal holes for beam tubes and through tubes. Part of this beryllium, near the fuel zone, is easily removable. When the effects of radiation damaging by high energy neutrons become significant and/or when the neutron poisoning of this zone by tritium — from (n, α) reactions — builds up to significant levels, this beryllium is replaced. The bulk of the beryllium reflector should endure for a number of years before it need be replaced for reasons of corrosion and/or radiation damage and/or buildup of tritium.

Between the beryllium and the pressure vessel, water is provided for shielding and to protect the pressure vessel against gamma and neutron radiation. Possibly, one or more neutron windows will be included in this zone, to permit placement of the bulk of the nuclear-instrumentation detectors outside the reactor vessel.

At the vertical level of the reactor core, outside the pressure vessel there is a zone of water, representing part of the reactor pool, and then a zone of magnetite concrete which provides the bulk of the lateral biological shield.

A summary of reactor parameters is given in Table I. Figures 3 and 4, respectively, show a vertical section and a horizontal section of the reactor vessel, and indicate locations of beam tubes and vertical irradiation tubes in the beryllium reflector. A typical beam tube assembly is shown in Fig. 5, and a horizontal section of the ITC is shown in Figure 6. Figure 7 presents a simplified flow sheet of the reactor. Beam tubes BT-7 and BT-8, shown in Fig. 4, are not included in the reference design. They are additive alternates in the design.

TABLE I

AARR Design Parameters

I.	Reactor Power Levels, MWt	
	A. Steady state, operating	100
	B. Minimum steady-state burnout	250
II.	Core Life at Full Power, days	90
III.	Calculated Thermal Neutron Fluxes at 100 MWt, 10^{15} n-cm/cm ³ -sec	

TABLE I
AARR Design Parameters (Contd.)

A. Maximum unperturbed in ITC	3.7
B. Maximum in typical ITC target ⁽¹⁾	2.7
C. Average in typical ITC target ⁽¹⁾	1.7
D. Maximum unperturbed in Be reflector	
1. Beginning of fuel cycle	1.0
2. End of fuel cycle	1.0
IV. Reactor Materials	
A. Fuel plate	Cermet of UO ₂ - stainless steel: stainless steel cladding
1. U ²³⁵ enrichment, %	93.
2. Total fuel loading of U ²³⁵ , kg	60.
3. Uranium oxide concentration in cermet, wt. %	37 ⁽²⁾
4. Total B ¹⁰ loading, g/fuel plate	0.04 (± 10%)
B. Coolant	H ₂ O
(1) Targets in a fully loaded ITC (assumed to be 75% water and 25% aluminum with additional neutron absorption corresponding to expected samples).	
(2) Graded to lower concentrations adjacent to ITC and near outer core boundary.	
C. Internal Thermal Column moderator	H ₂ O
D. External radial reflector	
Removable	Be + coolant H ₂ O
Permanent	Be + coolant H ₂ O
E. Shim, safety, and regulating control rods	
In-core: absorber section	Eu ₂ O ₃ -2TiO ₂
follower section	SS, or Zr-2
Peripheral: absorber section	Eu ₂ O ₃ -2TiO ₂
follower section	Zr-2

TABLE I

AARR Design Parameters (Contd.)

V. Heat Transfer and Coolant Data

A. General

1. System design pressure, psig	875
2. Normal operating pressure (vessel inlet), psig	750
3. Design coolant flow rates, gpm	
a. Through reactor vessel	24,660 (total)
1. Fuel-plate coolant channels	18,500
2. Control-rod channels	2,200
3. Beryllium reflector, including reflector irradiation facilities	3,070
4. ITC, including ITC irradiation facilities	330
5. Miscellaneous (beam tubes, etcetera)	560
b. Bypass flow and auxiliary primary system flows	2,540
[Total main pumping capacity, gpm	27,200]
4. System pressure drop, psi	175
5. Coolant temperatures (at 100 MWt), °F	
a. Vessel inlet	135
b. Vessel outlet	164
c. Maximum bulk water (fuel outlet)	250
d. Maximum surface (fuel plate-water interface)	378

B. Fuel region

1. Geometry	Flat plate
2. Fuel assembly dimensions, in.	
a. Height of active core	18
b. Total fuel plate height	20
c. Total fuel plate thickness	0.040

TABLE I

AARR Design Parameters (Contd.)

d. Cladding thickness (each surface)	0.005
e. Fuel-plate-core thickness	0.030
f. Coolant channel thickness	0.040
3. Number of fuel plates, total	1152
4. Total heat transfer area, ft ²	672.
5. Volume of active core, liters	78.
6. Power density, MWt/liter	
a. Maximum	5
b. Average	1.3
7. Heat flux, Btu/hr-ft ²	
a. Hot spot	1.9×10^6
b. Average	0.5×10^6
c. Burnout, at end of cycle, at operating pressure	4.9×10^6
8. Coolant flow rate, gpm	18,500
9. Coolant velocity, ft/sec	45
10. Total pressure drop along fuel element, psi	110
11. Temperature, °F	
a. Coolant at core inlet	135
b. Coolant average at core outlet	186
c. Clad-water interface (max.)	378
d. Clad-water interface (av.)	205
e. Fuel plate (max.)	707 (under spacer)
f. Fuel plate (av.)	270
g. Saturation temperature at hot spot	495

IV. Experimental Facilities

A. Permanent reflector

1. Vertical irradiation facilities

a. Total number	19
-----------------	----

TABLE I

AARR Design Parameters (Contd.)

- | | |
|-----------------------------|------------------------------|
| b. Number and diameter (ID) | 7 - 1-1/2 in. ⁽³⁾ |
| | 4 - 2 in. |
| | 6 - 2-1/2 in. |
| | 2 - 3 in. |

2. Horizontal tubes

- | | |
|--|--------------------------|
| a. Through tubes - number and diameter (ID) | 2 - 5 in. ⁽⁴⁾ |
| b. Blind beam tubes - number and diameter (ID) | 6 - 5 in. ⁽⁵⁾ |

-
- (3) Two 1/2 in. and two 1 in. hydraulic rabbit facilities are to be installed in four of the 1-1/2 in. vertical facilities.
- (4) Each through tube may be used as two blind tubes if desired.
- (5) Provisions are included for later addition of two blind beam tubes in a region of somewhat lower flux (BT-7 and BT-8, Figure 4).

B. Removable reflector

Vertical facilities	none ⁽⁶⁾
---------------------	---------------------

C. Internal thermal column

- | | |
|-------------------------|-------------|
| 1. Hydraulic rabbits | 5 - 1/2 in. |
| 2. Static facilities | 4 - 1/2 in. |
| 3. Gas-operated rabbits | (7) |

-
- (6) By installing special removable reflector elements having sample holes, a total of 24 is possible.
- (7) Provision for future addition of two facilities.

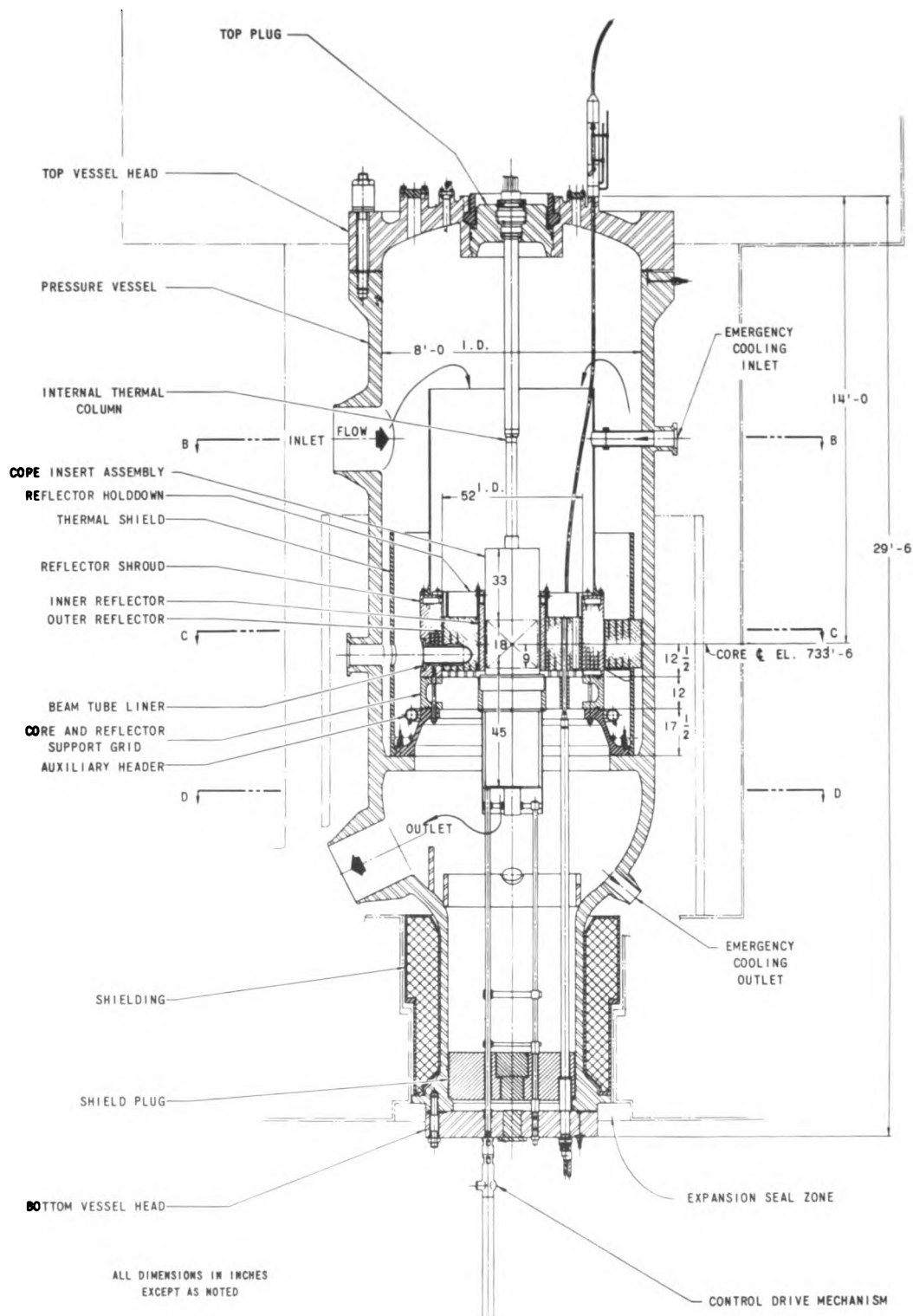


FIG. 3 VERTICAL SECTION OF REACTOR VESSEL

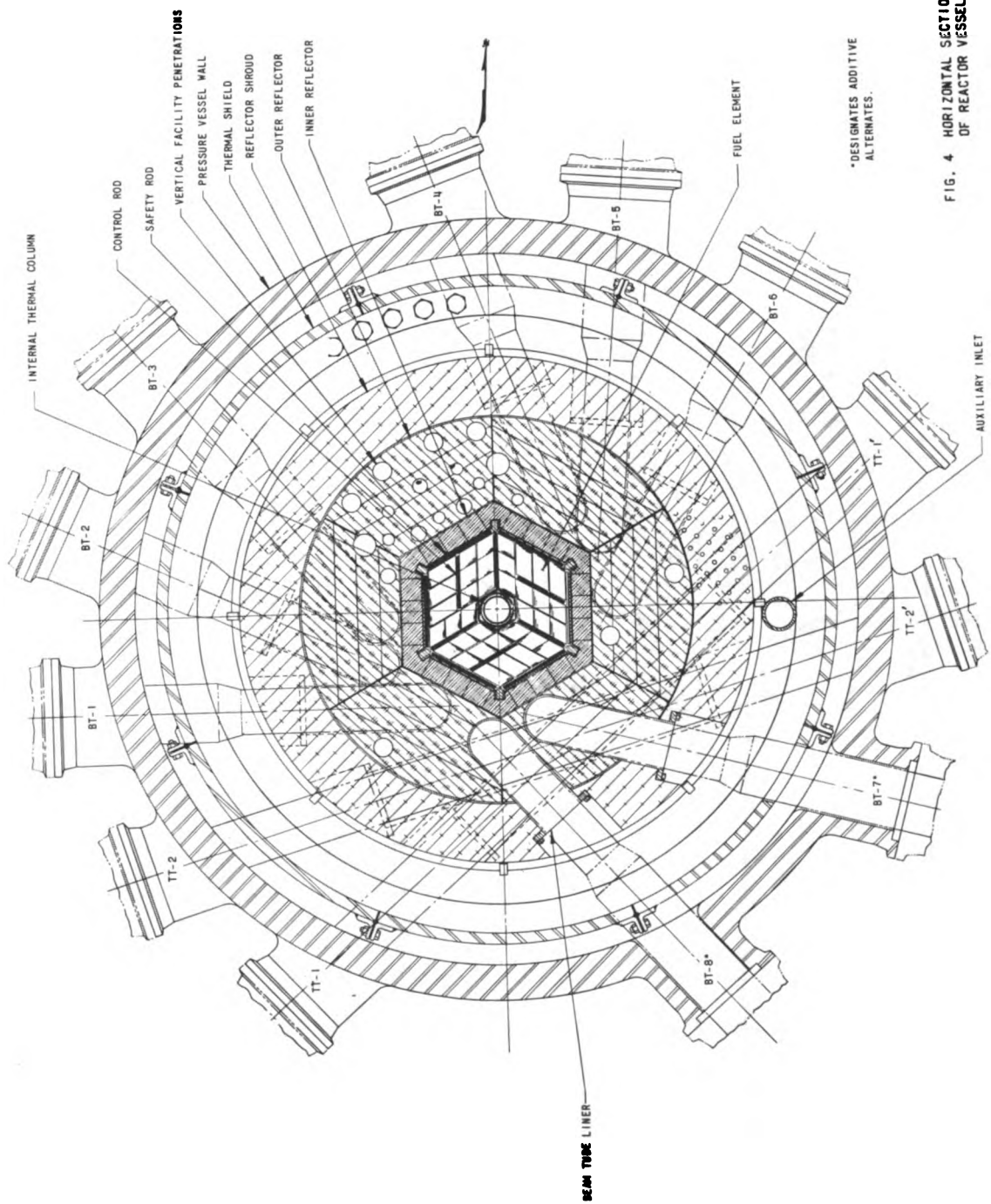


FIG. 4 HORIZONTAL SECTION OF REACTOR VESSEL

22-1

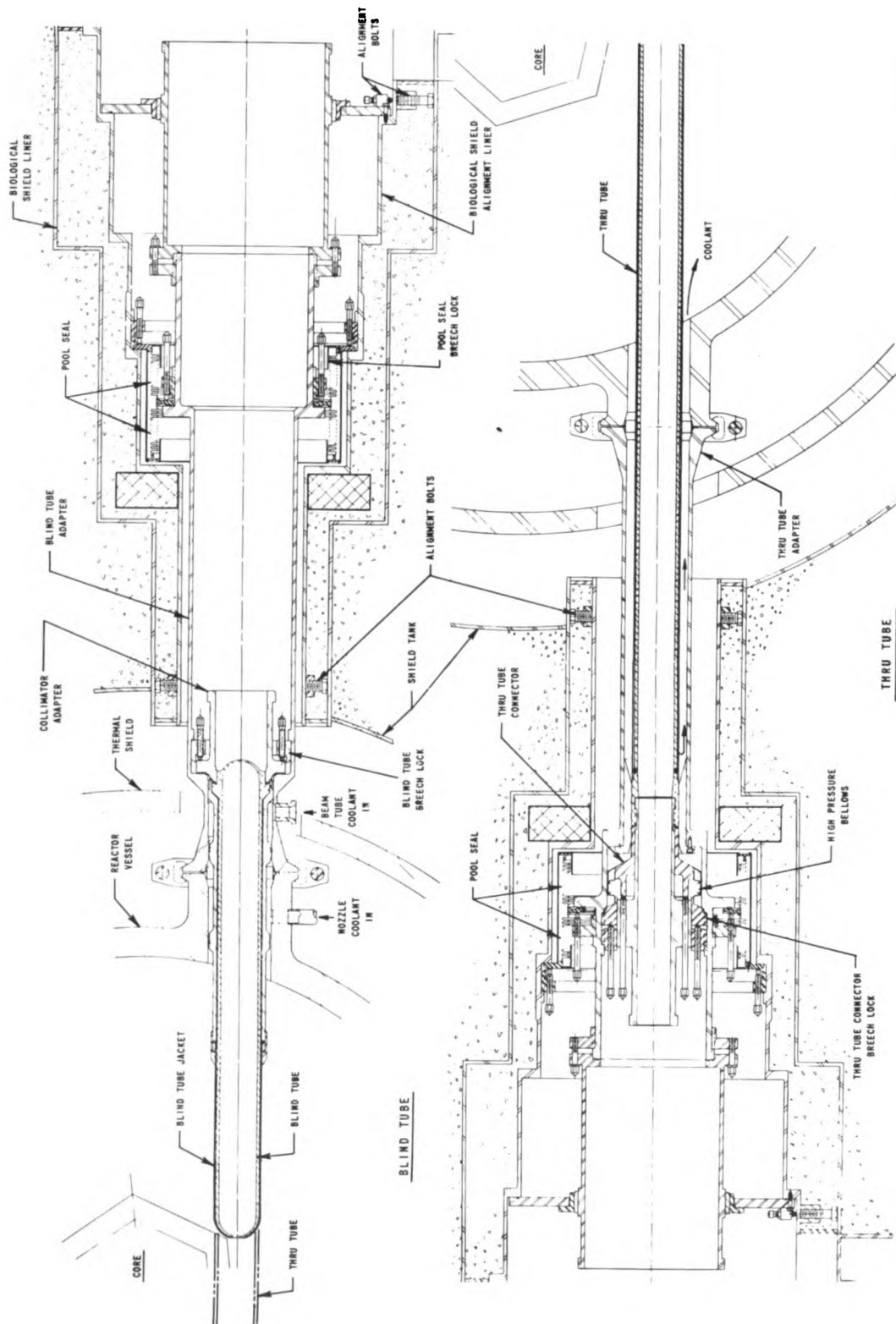


FIG. 5 BEAM TUBE ASSEMBLY

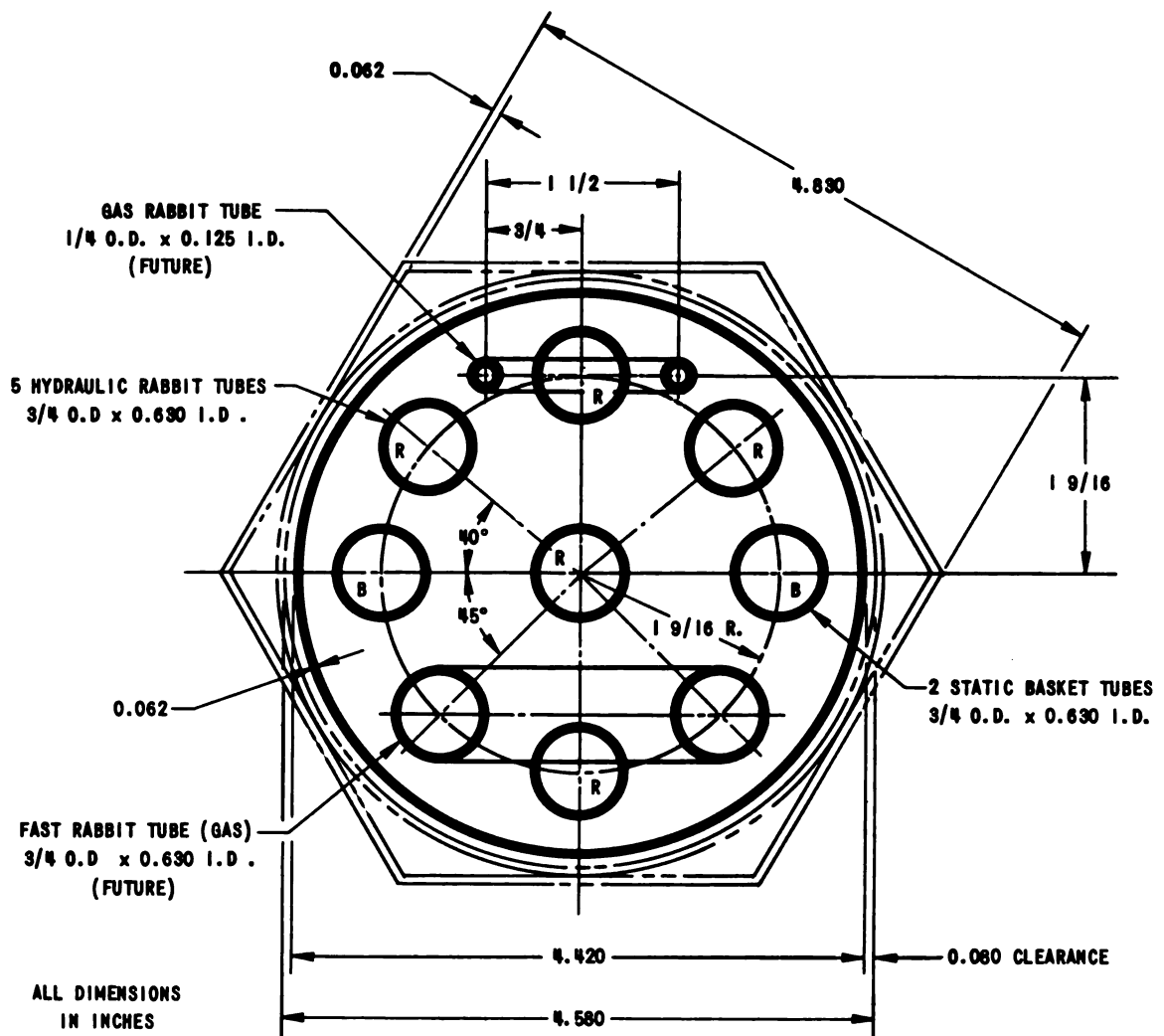


FIG. 6 HORIZONTAL SECTION OF THE INTERNAL THERMAL COLUMN

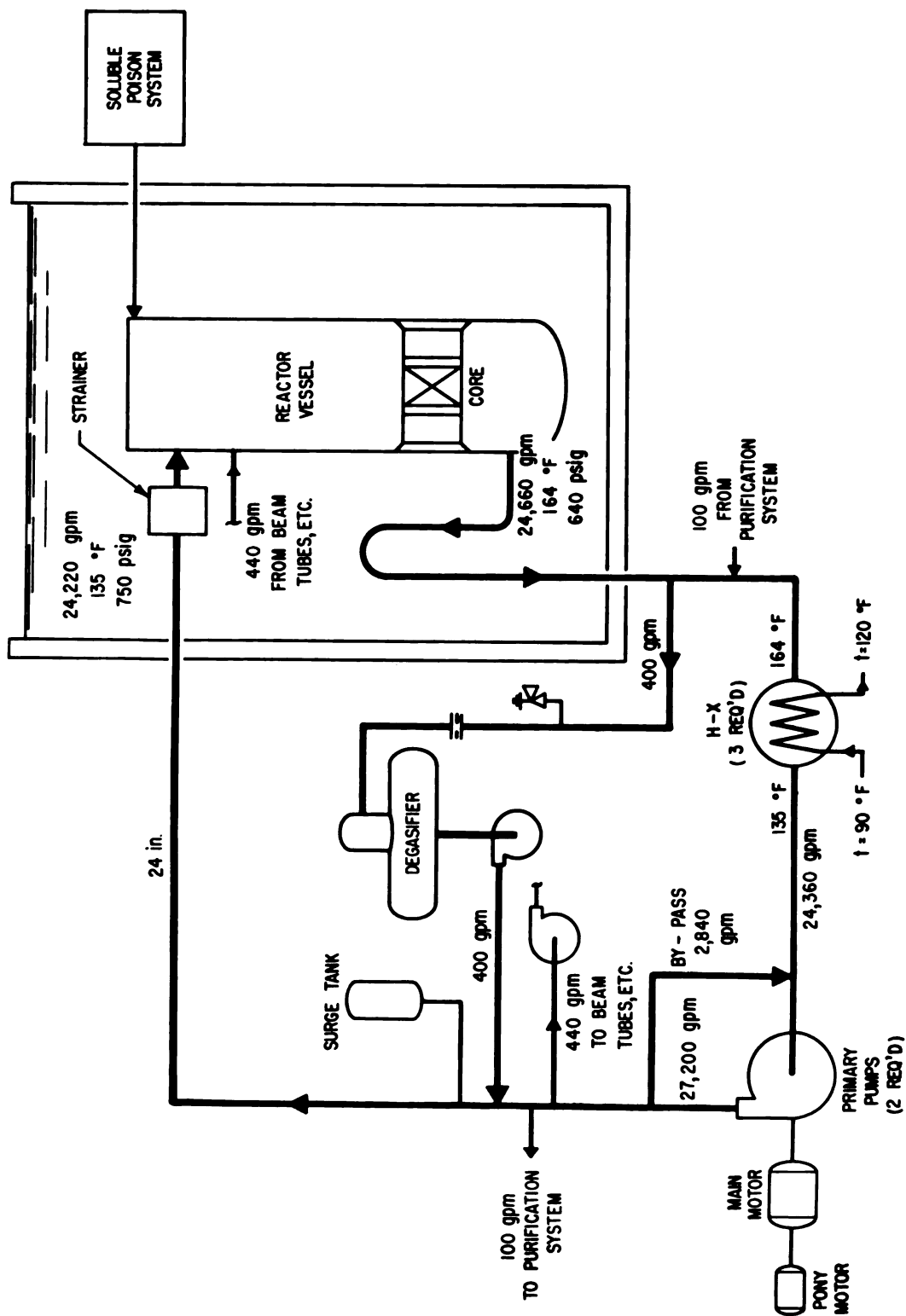


FIG. 7 AARR SIMPLIFIED FLOW SHEET

2. Summary of Costs

The construction budget for the AARR has been set at \$25 million, including the plant, the engineering design work of the Architect-Engineer firm, contingency funds, and \$1 million for experimenters' equipment.

At a reactor power level of 100 MWt, a reactor core life of 90 days is anticipated. In computing the costs of operating the plant on the basis of 24-hour/day usage at a full power level of 100 MWt, allowance was made for an average down time of 14 days between full core cycles. This assumption applies only to the cost accounting; it is our intent and our goal that the reactor will be at power for a larger fraction of the total time. On this basis, once the reactor is in routine, full-power operation at 100 MWt, it is expected that the annual cost of plant operation will be approximately $\$4\frac{1}{2}$ million. Of this, \$2 million per year is for costs other than those directly associated with fabrication (and reprocessing,...) of reactor cores and other zones of the reactor. The balance, $\$2\frac{1}{2}$ million per year, includes the costs of replacement of reactor cores, control rods, and portions of the beryllium reflector.

A total staff of approximately 110 persons is estimated, including service personnel, for the operation of this reactor — not including the experimenters or their assistants.

3. Summary Remarks on High-Flux Capabilities and Limitations of the AARR

To attain a long reactor core life at high power densities, the fuel content is maximized in the cermet of highly enriched UO_2 and stainless steel, and a large total heat-transfer surface is provided by using thin fuel plates and thin coolant gaps with a metal-to-water ratio near unity in the reactor core. The result is an undermoderated reactor core with a low-intermediate neutron energy spectrum in the bulk of the fuel zone. Important advantages accrue, including a core life estimated to be approximately 90 24-hr days at a reactor fission power level of 100 MWt. In such a system, with a median energy for fission well above the resonance region for neutron capture by Xe^{135} , the problem of overriding xenon upon shutdown is much less severe than in a thermal reactor. The capacity of the AARR to override maximum shutdown xenon during most of the core life offers a distinct advantage, both in the event of an unscheduled shutdown and in the event it is desired to shut down to check certain equipment or possibly to alter or replace experimental equipment.

These two advantages, namely, a long core life and xenon-override capability, are offset somewhat by a disadvantage that the annular fuel zone cannot be made so thin as one would like, to optimize the peak experimenters' fluxes per unit reactor power. The reactivity loss per cm reduction in fuel-

zone thickness is $\sim 1\frac{1}{2}\%$ in the fresh reactor, and the reactivity loss per 1000 MWd of energy production is increased as the core thickness is decreased. These reactivity losses would markedly reduce core life if the core thickness were reduced. With a thinner fuel zone, the peak thermal neutron flux in the inner flux-trap zone (ITC) per unit reactor power is increased. And so is the peak thermal neutron flux in the beryllium reflector outside the core. This occurs because: the probability of leakage of source neutrons is increased; the power density near these inner and outer flux-trap zones is increased; and the mean optical distance between the source and the receptor is decreased.

Of course, unless the ratio of the maximum power density to the average power density in the fuel zone is reduced appropriately, the effect of a reduction in fuel-zone thickness at fixed power is to increase the peak power density, thus reducing the burnout margin at the "hot spots" in the fuel plates. For the AARR, a substantial additional lowering of the maximum-to-average power in the fuel zone would require a more severe grading of the fissionable material (U^{235}), and extensions of the zones where such grading would be required. It has been calculated that this additional flattening of the gross radial power density in the AARR would have little effect on the experimenters' fluxes for fixed reactor power. However, there would be an additional loss in reactivity in the fresh reactor, and the loss in reactivity per 1000 MWd of operation would be greater.

For these reasons, it would be disadvantageous to reduce the thickness of the fuel zone substantially, and then possibly also flatten the power distribution much more. The penalty in core life would be severe. This is not to say that moderate reduction in fuel-zone thickness and/or moderate additional flattening of the gross radial power density could not be made to net advantage, with selected weightings for the algebraic worths of the increase in the peak experimenters' fluxes and the reduction in core life.

If the reactor power is increased much above 100 MWt, this disadvantage of the reference AARR core is diminished. It is conceivable that the final Mark-I (stainless steel) core selected for the AARR could be operated at power levels in excess of 200 MWt, without an increase in core size above the present reference size. At 200 MWt, the peak unperturbed thermal neutron flux in the ITC, computed for the reference AARR core, would be approximately 8×10^{15} n-cm/cm³-sec. The corresponding core life would be approximately 45 days at 200 MWt.

4. Reactor Physics Theory of the AARR *

A. Reactor Physics Bases for the Reactor Design

1. The Reactor Core

A heavily loaded annular reactor core has been selected for the reference Mark-I system. The fuel-plate matrix and cladding and most of the structural material in the reactor core is stainless steel. The steel provides a corrosion resistant medium with a high melting point. To some extent, it also provides a degree of reactivity control for the high U^{235} content of the core. An important feature of this heavily absorbing medium is that in the bulk of the reactor core the neutron flux spectrum may be characterized as low-intermediate. In the fresh reactor, the computed median energy of neutrons causing fission in this region is in the neutron energy range 10 eV - 30 eV. As the reactor is operated, the median neutron energy for fission decreases, but the reactor is still in the category of a low-intermediate system.

* A substantial experimental program has been carried out since October, 1964, in support of the reactor physics design of the AARR. Descriptions of the AARR Criticality Facility, and summaries of experimental data, have been published in Argonne's Reactor Development Program Monthly Progress Reports, and in Annual Reports of Argonne's Reactor Physics Division for 1964 (ANL-7010)⁽¹⁾, and for 1965 (ANL-7110).⁽²⁾ A report on a safety analysis of the Criticality Facility has been published.⁽³⁾ Papers have been presented at various meetings of the American Nuclear Society, including two papers on experimental data and theoretical analysis at the June, 1966 meeting in Denver, Colorado.^(4,5)

An important consequence of having an intermediate-spectrum fuel zone is that recovery of the reactor power to normal operating levels is not prevented by a rapid and massive buildup of xenon following a reactor shutdown. Power override of xenon is possible during most of the core life. Another important feature of this heavy fuel loading is that a substantial mass of U^{235} can be burned without too large a reactivity loss. The reactivity worth of fuel is estimated to be approximately - 2 1/2% for a reduction in fuel content from 60 kg U^{235} to 40 kg U^{235} . Thus, accounting also for the reactivity loss due to fission product absorption, a long core cycle is envisaged — perhaps three (3) months at 100 MWt.

The purpose of the AARR is to provide high-intensity fluxes of thermal neutrons, beyond the boundaries of the reactor core, for a multiplicity of basic research experiments. To fulfill this purpose, a thin annular reactor core has been chosen from which high-energy neutrons readily leak into radial internal and external reflectors where these neutrons are moderated to thermal energies. In the absence of flux depressors, or of a gradation in the loading of U^{235} from one fuel plate to the next, there would be very large power peaks

at the radial boundaries of the core. This is due to the steep gradients of thermal neutrons as they diffuse toward the heavily absorbing core from the internal and external reflectors. The present concept is that near radial core boundaries the weight percentage of UO_2 in the cermet will be varied to reduce the power peak. The gross radial maximum-to-average power ratio has been set, tentatively, at two (2). The net advantages of additional power flattening are under study.

The reference fuel plate is 0.1 cm (0.040 in.) thick, of which $\frac{3}{4}$ is a cermet of UO_2 and stainless steel. The nominal cladding thickness is 0.013 cm (0.005 in.) on each side. The thickness of the coolant channel has been set at 0.1 cm, also, even in the "hot" coolant channels, in the regions of radial power peaking. However, an evaluation is being made of the merits of using thicker coolant channels, e.g., 0.13 cm (0.050 in.), in these regions of higher power. The over-all ratio of water volume to non-water volume in the core is slightly less than unity. Therefore, the heavily absorbing reactor core is under-moderated, and the over-all reactivity coefficients of coolant void in the core, and thermal expansion of fuel plates, are negative.

As noted earlier, control of the available excess reactivity of the fresh reactor will reside in neutron poisoning by a combination of flat control blades and a burnable poison. The control-blade absorber will be Eu_2O_3 in a cermet with SS, stabilized by the addition of titania. From experiments performed in the AARR Criticality Facility, it is indicated that the six (6) in-core control blades selected will control a total of approximately $7\frac{1}{2}\%$ in reactivity. It is estimated that the six (6) peripheral control blades will control an additional $7\frac{1}{2}\%$ in reactivity.

The reference burnable neutron poison is boron, probably as a constituent of the fuel-plate cores. The estimated excess reactivity to be assigned to control by burnable poison is $\sim 6\%$.

2. The Internal Thermal Column (ITC)

Extensive reactor physics studies, both computational and experimental, have amply demonstrated that a moderating zone surrounded by a neutron-source medium provides an effective flux trap for thermalized neutrons. Ordinary water (H_2O) has been chosen for the ITC of the AARR for two basic reasons: (1) the moderator/coolant selected for the reactor fuel zone is H_2O , and the reactor design would be complicated greatly by the use of a different fluid for the ITC; and (2) of the various mixtures of H_2O and solid non-hydrogenous moderators (graphite, beryllium) the computed peak thermal neutron flux in feasible systems is attained with 100% H_2O . However, with water as the moderator in the ITC, the

reactivity coefficient of partial voids in the ITC is positive. Therefore, consideration is being given to the use of a solid hydrogenous moderator, or beryllium, in place of some of the non-coolant water in the ITC. The advantage here would be that if some or all of the water in the ITC were removed (expelled,...) the maximum reactivity gain would be much smaller — a safety advantage.

The volume of the ITC has been set at ~ 7 liters. This corresponds to the volume of a right-circular cylinder of height 45.7 cm and radius ~ 7 cm. This large radius was selected to allow for space for additional target tubes, and anticipate that heavier loadings or absorbing samples might be wanted in the ITC at a later date. In the unperturbed (100% H_2O) ITC, the computed value of the peak thermal neutron flux per unit core power is almost independent of ITC radius in the range from 6 cm to 7 cm radius. The net effects of reducing the radius of the ITC to less than the reference 7-cm radius are being evaluated. One advantage of the smaller ITC is that the maximum positive reactivity effect of partial voiding of the ITC is smaller. An obvious disadvantage of the smaller ITC is the reduction in the area available for vertical irradiation facilities. Also, if the ITC were loaded with target tubes containing samples, the average intensity of the thermal neutron flux would be lower in the smaller ITC. If the larger (reference) ITC were loaded very heavily, which is not intended at this time, the peak reactivity gain due to voids in the ITC would be dramatically reduced.

As determined both by experiment and by computation, the reactivity coefficient for partial voiding of the ITC is positive. The overriding constraint of the detailed ITC design, and the constraint of the samples to be inserted, is the minimization of the possibility of a massive partial voiding of the ITC.

3. The Radial Beryllium Reflector

For the external radial reflector, beryllium has been chosen as the moderator, rather than graphite or H_2O , to achieve high thermal neutron fluxes in beam tubes penetrating this zone, and, at the same time, to design for the location of vertical irradiation facilities in regions of high thermal neutron flux. It is expected that the beryllium reflector will be at least 1 ft thick. In the computations, and in the critical experiments, a thickness of 1 ft has been assigned. The first 2 to 3 in. of beryllium is a "temporary" zone, which will be easily replaced by new beryllium if such replacement is considered to be desirable before beginning a given new core cycle. The remainder of the beryllium is considered to be "permanent", in the sense that it will be left in place for a number of years.

The use of a full H_2O reflector, or of a thin beryllium zone followed by H_2O , would cause too rapid a drop off in the level of the thermal neutron flux beyond the first few inches of reflector. This would greatly limit the high-flux zone for vertical irradiation facilities in the reflector. In addition, the beryllium reflector provides a substantial bonus of reactivity. It has been computed that, for a 10-in.-thick permanent beryllium zone, removal of this beryllium, and its replacement by water, would cause a reactivity loss of approximately 2%. For the case of a 2-in.-thick temporary beryllium zone, if, in addition, the temporary beryllium were replaced by water, there would be an additional reactivity loss of at least 5%. The magnitude of the latter reactivity loss depends upon the height of the beryllium. The use of a beryllium zone which extends quite some distance beyond the axial boundaries of the active fuel zone would increase this magnitude.

Aluminum-walled beam tubes will penetrate the radial beryllium reflector to within a few inches of the reactor core. Possibly, local neutron-flux traps (H_2O) will be positioned around the tips of the "blind" beam tubes. The effects of such flux traps on reactivity and on the levels of thermal neutron flux at the tips of the beam tubes — and in the beam tubes — will be measured in simulated tubes in the AARR Criticality Facility. Aluminum-walled through tubes will penetrate the beryllium, but at a greater distance from the reactor core.

The reactivity coefficient of voiding of reflector coolant water has been computed to be negligibly small, and of indeterminate sign, in the absence of substantial H_2O flux-trap zones near the reactor core. As detailed design of the external radial reflector proceeds, special attention will be paid to the reactivity effects of voids in flux traps in this zone and of flooding of beam tubes and through tubes.

B. Methods of Reactor Physics Analysis

The bulk of the reactor physics computations summarized here have been multigroup diffusion theory (REX)⁽⁶⁾ computations in one-dimensional, cylindrical geometry. A total equivalent bare core height of 60 cm was assigned. The basic 16-group partition of the neutron energy range is that of G. E. Hansen and W. H. Roach.⁽⁷⁾ The energy and lethargy partitions are listed in Table II.

The basic Hansen-and-Roach library of cross sections was used for groups where the neutron energy was greater than 0.4 eV (groups Nos. 1 to 14, inclusive). Typically, for those materials which were not included in the Hansen-and-Roach library, group-average cross sections were obtained from GAM-1⁽⁸⁾ computations. Absorption cross sections for nickel at high energies were estimated independently.

TABLE II

Multigroup Partition of the Neutron Energy Range

<u>No. of Neutron Energy Group</u>	<u>Neutron-Energy Range</u>	<u>Lethargy Width, Δu</u>	<u>Normalized Fission- Spectrum Partition</u>
1	3 - ∞ MeV		0.204
2	1.4 - 3 MeV	0.762	0.344
3	0.9 - 1.4 MeV	0.442	0.168
4	0.4 - 0.9 MeV	0.811	0.180
5	0.1 - 0.4 MeV	1.386	0.090
6	17 - 100 keV	1.772	0.014
7	3 - 17 keV	1.735	0
8	0.55 - 3 keV	1.696	0
9	100 - 550 eV	1.705	0
10	30 - 100 eV	1.204	0
11	10 - 30 eV	1.099	0
12	3 - 10 eV	1.204	0
13	1 - 3 eV	1.099	0
14	0.4 - 1 eV	0.916	0
15	0.1 - 0.4 eV	1.386	0
16	0 - 0.1 eV		0

The material group-average cross sections for the two thermal neutron energy groups, group No. 15 (0.1 eV to 0.4 eV) and group No. 16 (0 to 0.1 eV), were obtained from calculations with the transport theory code THERMOS,⁽⁹⁾ using the Nelkin scattering kernel for water and the free-gas kernel for other materials. Cross sections for net downscattering of neutrons from group No. 15 to group No. 16 were inferred from the output neutron flux spectra of THERMOS for each of the various zones of the reactor. With these approximations for the net effects of energy exchange between neutron and scatterer, it was possible to avoid inclusion of up-scattering in the REX computations. The Argonne version of the THERMOS program has been revised to calculate the full scattering matrix (upscatter and downscatter), isotope by isotope, for as many as 48 speed groups.⁽¹⁰⁾ Calculations are being made now with two thermal groups, including upscattering, but the data reported in this paper are based entirely on calculations involving only net downscattering.

The cross section for net downscatter by beryllium, from group No. 15 to group No. 16, is particularly sensitive to the effects of localized neutron leakage. Therefore, but within the range of uncertainty of this parameter, a reasonable value for this parameter was selected on the basis of comparisons of computed and measured activation distributions of neutron detectors in the pure beryllium reflector of the AARR Criticality Facility. This is discussed in greater detail in Reference 2. In that report are shown comparisons of measured and calculated radial distributions of neutron-detector activity. With the use of the modified $\sigma_{s_{\text{net}}}^{\text{Be}}$ (15→16), agreement between theory and experiments was satisfactory where such comparison reasonably could be made, namely, away from reactor core boundaries.

In these one-dimensional calculations the radii of the various annular regions were chosen to yield the cross sectional area of the actual zones. The materials in each of the various zones of the reactor were assumed to be effectively homogeneously distributed. The design-reference system upon which these calculations are based is summarized in Table III. The detailed representation of the grading of fuel in the fuel-zone annulus is given in Table IV.

TABLE III

Radii of Equivalent Cylindrical Annuli

Region Composition (Volume Fractions)	Outer Radius, cm
ITC water	6.63
ITC liner (SS) ^(a)	6.83
Fuel zone ^(b)	24.26
H ₂ O 0.4754	
UO ₂ 0.1073 ^(c)	
SS 0.4173	
Control-blade follower region	25.29
Zr 0.715	
H ₂ O 0.285	
Beryllium reflector region, Zone No. 1	30.89
Be 0.90	
H ₂ O 0.10	
Beryllium reflector region, Zone No. 2	43.09
Be 0.810	
H ₂ O 0.070	
Al 0.050	
void 0.070	
Beryllium reflector region, Zone No. 3	55.79
Be 0.794	
H ₂ O 0.050	
Al 0.056	
void 0.100	
Water zone	121.92
H ₂ O 0.850	
Al 0.050	
void 0.100	

(a) In the final design, it is likely that a zircaloy will be used for this liner.

(b) This is the composition of the ungraded region of the core. The fuel content of the fuel plates at the inner and outer edges of the core has been reduced. For the detailed fuel loading, see Table IV.

(c) A fully enriched (93.2% U²³⁵) UO₂-SS cermet is assumed. The density of the oxide is taken to be 9.3 g/cm³.

TABLE IV

Graded Loading of U^{235} in the Reference Design Core

No. of Fuel-Zone Annulus	Radial Dimensions of Annulus, cm	Fractional Loading ^(a) of U^{235}
1	6.83 - 7.02	0.22
2	7.02 - 7.20	0.28
3	7.20 - 7.38	0.37
4	7.38 - 7.56	0.47
5	7.56 - 7.73	0.60
6	7.73 - 7.90	0.75
7	7.90 - 8.06	0.91
8	8.06 - 23.67	1.00
9	23.67 - 23.87	0.78
10	23.87 - 24.06	0.57
11	24.06 - 24.26	0.42

(a) The fractional loading is the ratio of the U^{235} content in this zone to the U^{235} content in the bulk of the reactor where the fuel-plate core contains 37 wt % highly enriched UO_2 .

C. Calculated Reactor Physics Parameters

The total available excess reactivity of the fresh unpoisoned reactor is calculated to be $\sim 17\%$ at operating temperature. From this must be subtracted the reactivity losses due to: (1) "equilibrium" xenon and samarium ($\sim 3\%$, at 100 MWt); (2) additional reductions in the reactivity worth of the beryllium reflector, due to: local concentrations of absorption in the vertical irradiation facilities; streaming of neutrons in the beam tubes; and inclusion of water for neutron flux trapping near the tips of the "blind" beam tubes; (3) additional localized reductions in U^{235} content in certain fuel plates near the inner and outer radial boundaries of the reactor core, or other reactivity losses of this type; and (4) allowance for the possibility that the ITC might be used more nearly as an isotope-production medium at some time. Also, from comparisons of calculated and measured excess reactivity for the various reactor core loadings, it is indicated that calculation might over-estimate reactivity by $\sim 1\%$. Of the remaining reactivity, $3\%-4\%$ will be designated as an operating margin, to be controlled by the in-core control blades. It is intended that the remaining 6% or 7% in reactivity will be controlled by a burnable poison. The current choice is boron, probably as a constituent of the fuel-plate cores. For a uniform spatial distribution in the core, the computed average reactivity worth of boron is -0.024% per gram of natural boron.

From measurements in the Criticality Facility,^(2,11) it is estimated that the total reactivity control capacity of the 12 control blades will be approximately 15% . This is adequate to control the net reactivity not controlled by burnable poison in the fresh reactor at room temperature, even if one control blade were stuck in its position of full withdrawal. An auxiliary (backup) means of reactor control will be provided in the form of a solution of a boron compound which could be introduced into the reactor water. Such control would be introduced relatively slowly, as a supplement to the reactivity control by the control blades, if necessary. A uniform poisoning of the reactor water to a content of 0.6 g natural boron per liter would reduce reactivity by 2% . For control of 15% reactivity, a concentration of ~ 10 g B/liter of water would be needed.⁽¹²⁾

The calculated value of β_{eff} in the preliminary reference reactor is 0.0072 . A perturbation-theory calculation of the mean prompt neutron lifetime has yielded a value of ~ 22 microseconds; this appears to be reasonable, in comparison with calculations and Rossi- α measurements on systems in the Criticality Facility.^(2,13)

Starting with the fresh preliminary reference reactor, at room temperature, it has been calculated that approximately $\frac{1}{2}\%$ in reactivity is lost in going to the temperatures corresponding to operation at 100 MWt. This

includes the temperature coefficients of reactivity in the internal and external radial reflectors. The principal reactivity contributions in the reactor core are due to the corresponding changes in the density of the water. The calculated coefficient of reactivity due to a change in density of water in the fresh reactor core is -0.23% per % void in core water.

Equilibrium xenon, at 100 MWt, controls approximately 2% in reactivity, and equilibrium samarium controls approximately $\frac{3}{4}\%$.

The reactivity coefficient for small-fraction void in the unperturbed 7-cm ITC has been computed to be approximately $+0.04\%$ per % ITC void. The measured void coefficient of reactivity is $\sim 0.04\%/ \%$ void, in the existing AARR Criticality Facility ITC, which has an equivalent-circle radius of ~ 6 cm. The corresponding computed void coefficients of reactivity for the smaller ITC are roughly $1/4$ smaller. These computations have been made in one-dimensional geometry, with no corrections for the increased effectiveness of the axial reflectors — a positive reactivity effect. Two-dimensional computations are in progress, using the 2-D diffusion theory program CANDID.⁽¹⁴⁾

D. Neutron Flux Spectra and Power Distributions

In the reactor fuel zone, a useful characterization of the neutron flux spectrum is the fractional contribution of fission-spectrum neutrons from each of the 16 neutron energy groups. This spectral indexing, calculated for the ungraded (bulk) portion of the fuel zone, is given in Table V.

Figures 8-10 include relative radial distributions of neutron flux for an ITC without target tubes. These curves are all normalized to a value of unity for the group-16 flux at the axis or the ITC. The calculated absolute value for the group 16 flux (0 - 0.1 eV) at the center of the ITC is 3.0×10^{15} at 100 MWt. In these calculations, the effects of a design involving local flux traps at the tips of the blind beam tubes have not been included.

TABLE V

Fractional Contribution of Each Neutron-Energy Group to
Production Rate of Fission-Spectrum Neutrons^(a)

No. of Neutron-Energy Group	Minimum Neutron Energy in Group	Fractional Contribution ^(b,c)
1	3 MeV	0.014
2	1.4 MeV	0.027
3	0.9 MeV	0.015
4	0.4 MeV	0.022
5	0.1 MeV	0.026
6	17 keV	0.032
7	3 keV	0.040
8	0.55 keV	0.065
9	100 eV	0.128
10	30 eV	0.099
11	10 eV	0.102
12	3 eV	0.078
13	1 eV	0.049
14	0.4 eV	0.066
15	0.1 eV	0.138
16	0 eV	0.099

(a) This fractional contribution is calculated in the bulk of the reactor core, where the U^{235} loading is highest, at a radial position of ~ 15 centimeters.

(b) The fractional contribution as $(\nu\Sigma_f)_j\phi_j/\sum_k(\nu\Sigma_f)_k\phi_k$, where Σ_{fj} is the macroscopic fission cross section in group No. j , and ϕ_j is the group- j neutron flux as printed out following the multigroup computation in cylindrical geometry.

(c) The sum of the fractional contributions is slightly larger than unity, due to round-off. Although these fractions are listed to three decimal places, as computed, it must be understood that the spectrum of fission events is sensitive to the particular model employed in computing group cross sections.

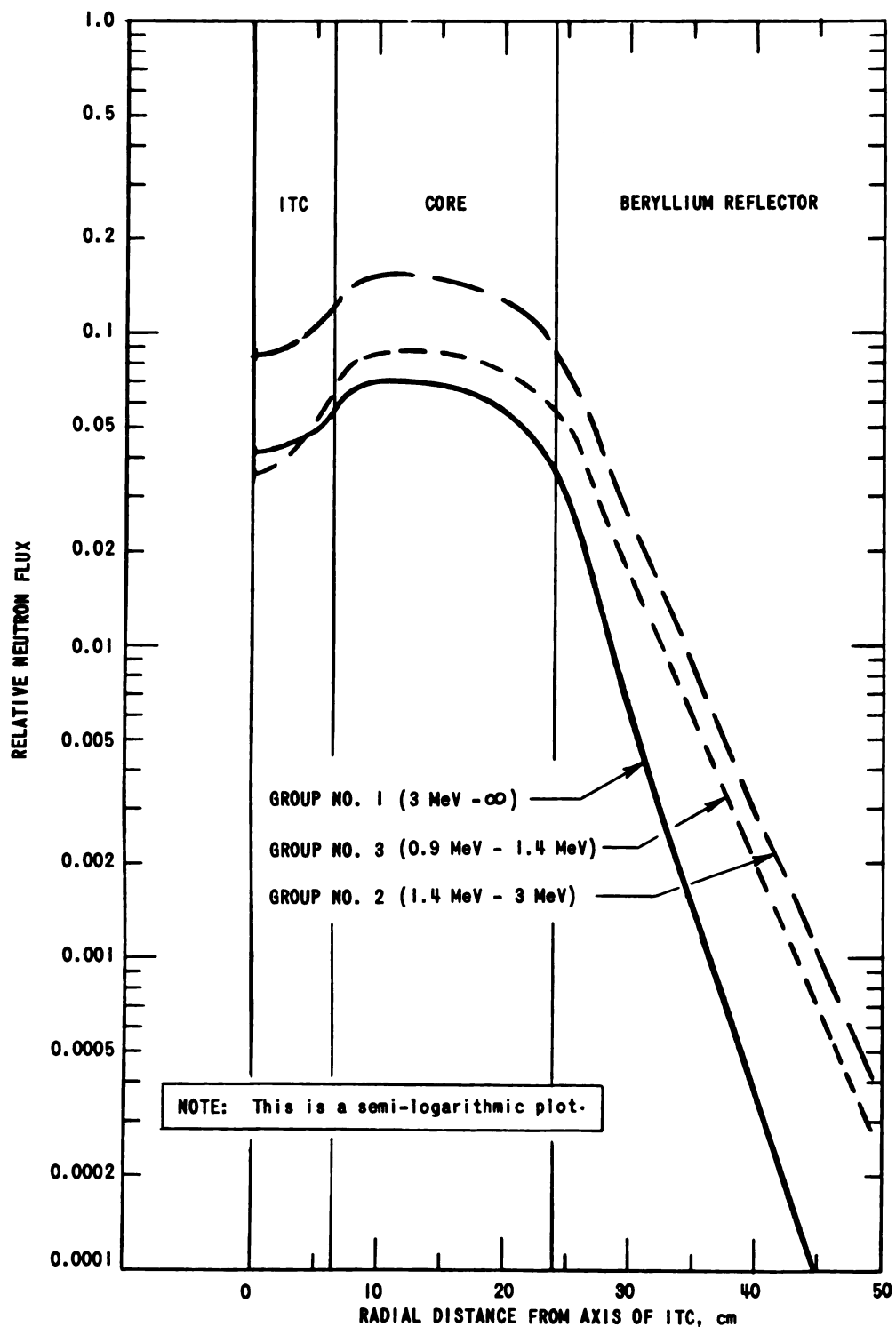


FIG. 8 RADIAL DISTRIBUTIONS OF NEUTRON FLUX IN THE REFERENCE DESIGN AARR
(NORMALIZED TO UNIT GROUP-16 FLUX AT CENTER OF INTERNAL THERMAL COLUMN)

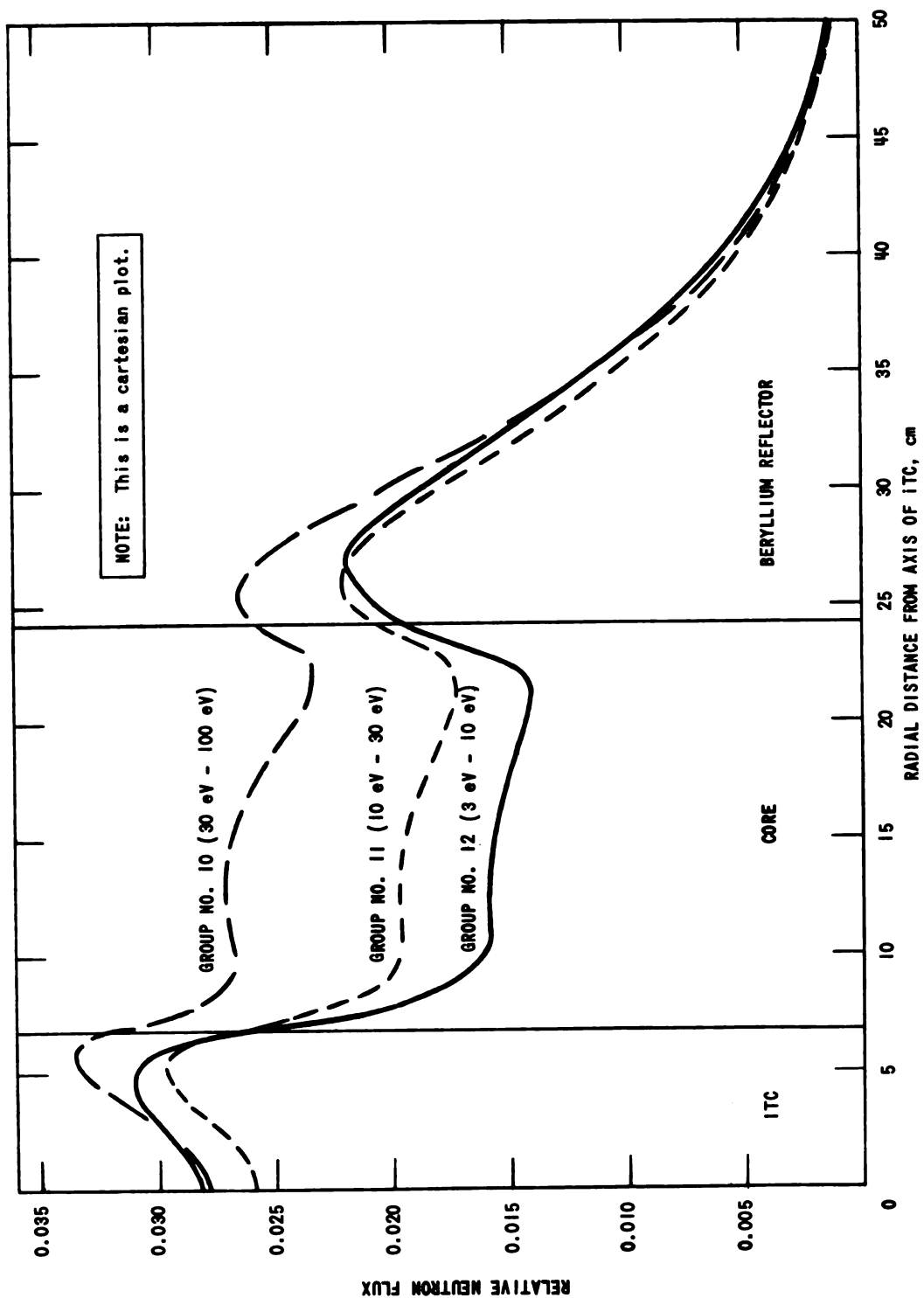


FIG. 9 RADIAL DISTRIBUTIONS OF NEUTRON FLUX IN THE REFERENCE DESIGN AARR
(NORMALIZED TO UNIT GROUP-16 FLUX AT CENTER OF INTERNAL THERMAL COLUMN)

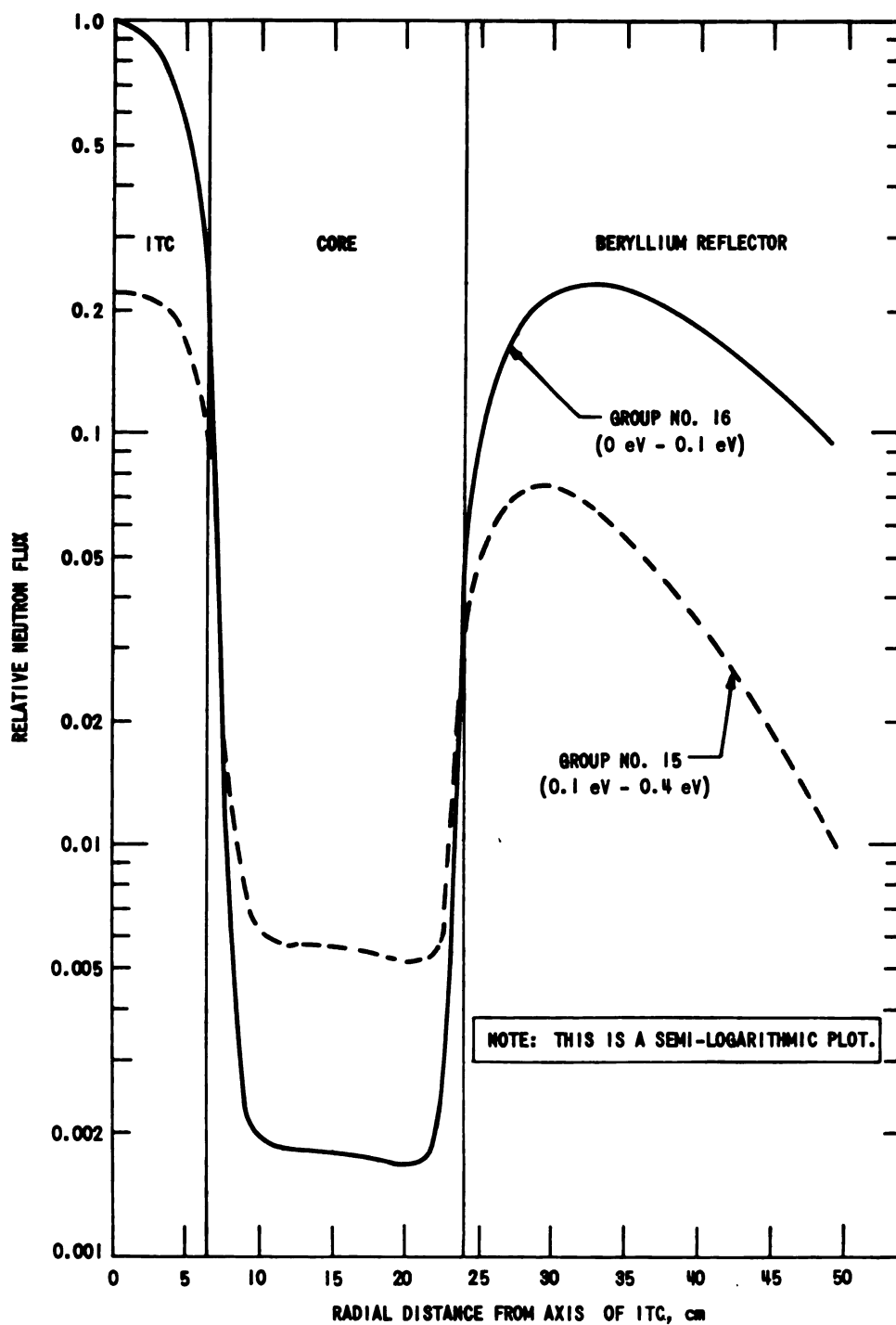


FIG. 10 RADIAL DISTRIBUTIONS OF NEUTRON FLUX IN THE REFERENCE DESIGN AARR
(NORMALIZED TO UNIT GROUP-16 FLUX AT CENTER OF INTERNAL THERMAL COLUMN)

Figure 11 shows the calculated gross radial distribution of the fission-neutron production rate per unit volume of core. Each of the graded-fuel zones is approximated as the volume-averaged composition of the single fuel plate and its associated coolant zone. This distribution is an excellent approximation to the corresponding calculated distribution of power, since $\bar{\nu}$ (neutrons per fission) is almost constant throughout the core.

Table VI gives more detailed information about the neutron flux spectrum in a horizontal plane at the midheight of the reactor core. In Table VI, at each radial position, the neutron flux spectrum is normalized to unity for the sum over the entire 16 neutron energy groups. The hardening of the spectrum at the ITC-core interface and at the core-beryllium interface is clearly evident.

Unless otherwise stated, all of the data apply to the fresh, unburned reactor. Figure 12, however, shows the variation of the group-16 neutron flux, in the ITC and in the beryllium reflector, as the core is operated. Note that the calculated time-dependence of the flux is small. These calculations, in one-dimensional (cylindrical) geometry, do not account for the spatial perturbations caused by control rods, to the extent that spatially non-uniform burnup of the fuel causes such perturbations.

The sample content of the Internal Thermal Column affects the flux levels in the ITC. Table VII lists calculated data of relative thermal neutron flux for aluminum contents ranging from zero to 50% of the ITC volume, and for total target absorption areas ranging from zero to the area provided by 1.0 g of natural boron in the 18-in.-high "active" portion of the ITC. Approximately, 1.0 g boron corresponds to an absorption area of 40 cm², in the group-16 energy range, and an absorption area of 20 cm², in the group-15 range. It is anticipated that the fully loaded ITC will contain approximately 25% aluminum and that the target absorption area will be less than $\frac{1}{2}$ g-boron equivalent.

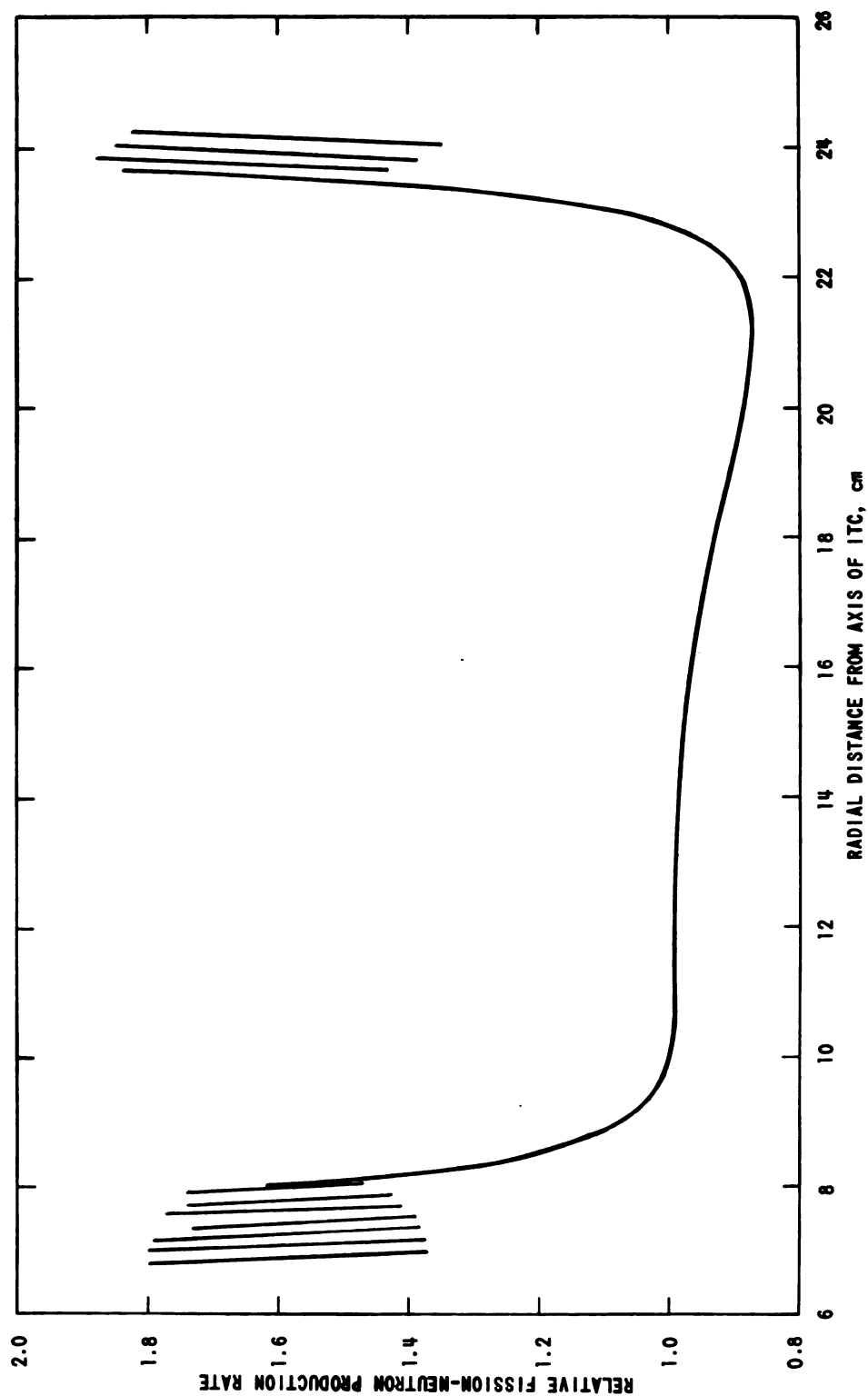
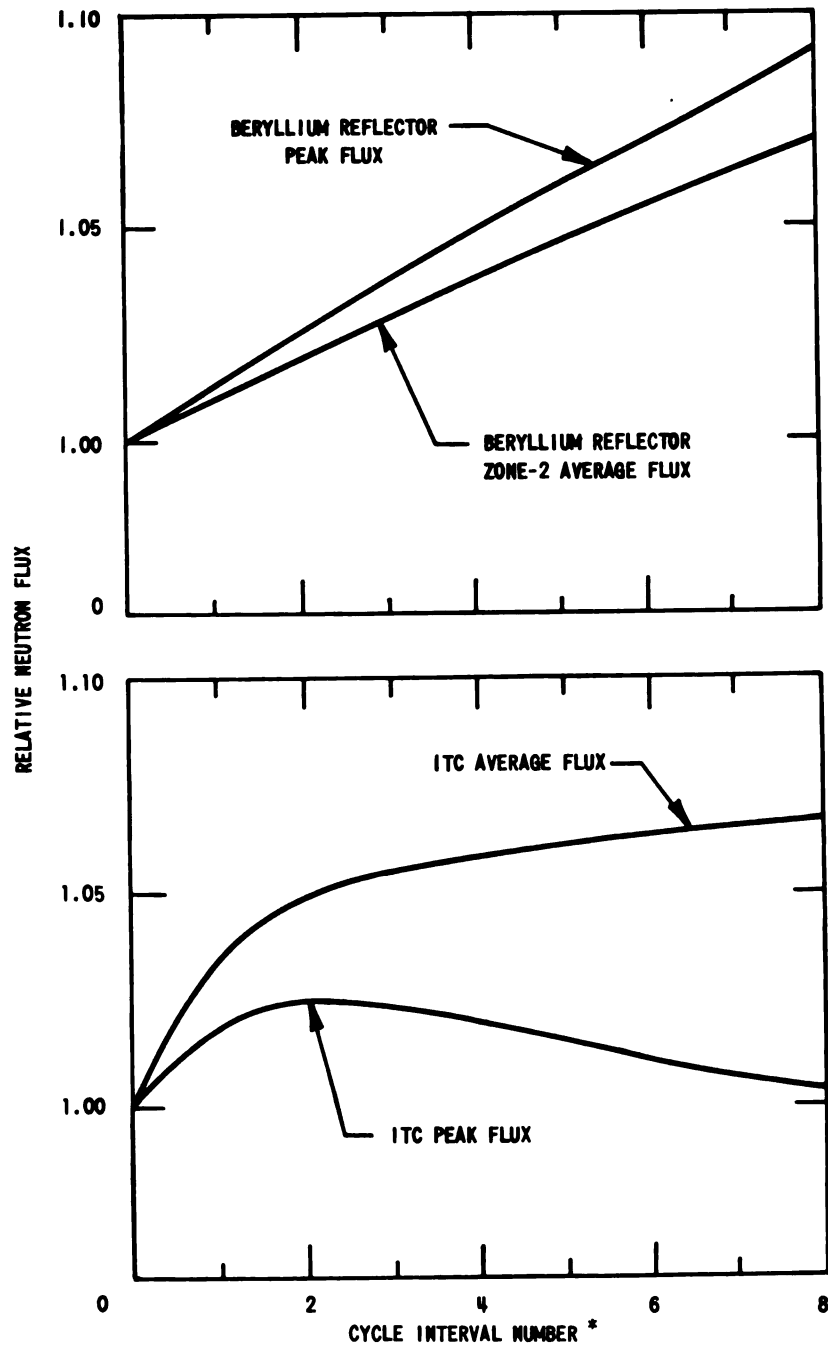


FIG. 11 RADIAL DISTRIBUTION OF FISSION-NEUTRON PRODUCTION RATE
(NORMALIZED TO UNITY AT 10.0 cm)

TABLE VI
Normalized Neutron Flux Spectrum in the ITC and Beryllium Reflector^a

Number of Neutron Energy Group	Minimum Neutron Energy in Group	Neutron Spectrum in the ITC										Neutron Spectrum in the Beryllium Reflector			
		Average over the ITC Region													
		r = 0	r = 2.0 cm	r = 4.0 cm	r = 6.6 cm							r = 25.3 cm	r = 29.0 cm	r = 32.9 cm	r = 44.2 cm
1	3 MeV	0.023	0.024	0.029	0.054	0.033	0.037	0.013	0.005	0.0006					
2	1.4 MeV	0.047	0.050	0.061	0.116	0.070	0.098	0.049	0.027	0.006					
3	0.9 MeV	0.020	0.022	0.028	0.065	0.034	0.063	0.031	0.017	0.004					
4	0.4 MeV	0.030	0.032	0.043	0.097	0.051	0.101	0.055	0.032	0.009					
5	0.1 MeV	0.033	0.036	0.046	0.102	0.056	0.114	0.075	0.049	0.015					
6	17 keV	0.025	0.028	0.036	0.075	0.042	0.086	0.064	0.046	0.016					
7	3 keV	0.022	0.023	0.030	0.057	0.034	0.068	0.055	0.042	0.017					
8	0.55 keV	0.021	0.022	0.028	0.052	0.032	0.060	0.050	0.040	0.018					
9	100 eV	0.022	0.023	0.028	0.049	0.032	0.055	0.049	0.040	0.019					
10	30 eV	0.016	0.017	0.020	0.032	0.022	0.034	0.033	0.028	0.015					
11	10 eV	0.014	0.015	0.018	0.027	0.020	0.028	0.029	0.026	0.014					
12	3 eV	0.016	0.017	0.019	0.027	0.021	0.027	0.030	0.028	0.016					
13	1 eV	0.015	0.015	0.018	0.023	0.018	0.022	0.026	0.025	0.015					
14	0.4 eV	0.012	0.013	0.014	0.017	0.015	0.017	0.021	0.020	0.013					
15	0.1 eV	0.124	0.125	0.124	0.074	0.115	0.061	0.111	0.125	0.104					
16	0	0.560	0.537	0.458	0.132	0.406	0.129	0.309	0.449	0.718					

^aAt each radial position, r , the fluxes are normalized to a total flux of unity, summed over all 16 energy groups.



* A SINGLE CYCLE INTERVAL REPRESENTS 1500 MWD

FIG. 12 VARIATION OF THE PEAK AND AVERAGE GROUP-16 (0 TO 0.1 eV) FLUXES IN THE ITC AND BERYLLIUM REFLECTOR WITH CORE BURNUP

TABLE VII

Relative Thermal Neutron Fluxes in Loaded ITC's^a

Target Absorption, ^b g (natural) boron	Volume-Percent Aluminum in ITC					
	0		25		50	
	Peak	Average	Peak	Average	Peak	Average
0	1.00	1.00	0.80	0.79	0.49	0.50
0.5	0.91	0.92	0.73	0.73	0.46	0.47
1.0	0.84	0.86	0.68	0.69	0.43	0.45

^aThe "thermal neutron flux" is defined as the sum of the fluxes in groups Nos. 15 and 16. The relative peak fluxes are normalized to unity for the case of no target tubes (100%-H₂O ITC); the relative average fluxes are normalized similarly.

^bThis absorption area of the target is in addition to the absorption area of the aluminum diluent.

E. Effects of Variations in Reactor Core Parameters

Calculations have been made of the reactivity effects of varying: (1) the equivalent-bare-core height; (2) the ratio of water/non-water in the core; (3) the thickness of the ungraded fuel zone; and (4) the fuel loading.

For variation No. 1: a reduction of 4 cm in height would reduce reactivity by 1.6%; an increase of 4 cm in height would increase reactivity by 1.3%. The curve of reactivity versus height is monotone increasing, with increasing height, and it is convex upward in shape. These calculations were made for the design-reference system.

For variation No. 2, the calculations were performed for an ungraded fuel loading at room temperature. The results, given in Fig. 13, are for two types of variations. In Case A, the total mass of UO₂ is fixed; as the water content is varied, the volume of SS is varied to maintain constant total core volume. In Case B, the ratio of SS to UO₂ is held constant, at the value for the design-reference AARR. In each case, the curve of reactivity versus the volume ratio water/non-water is monotone increasing, and convex upward in shape, for volume ratios in the range 0.5 to 1.5. Note that such a change in metal content would mean a change in the heat transfer area, and this has been an important factor in the selection of the ratio of SS/H₂O in the reactor core.

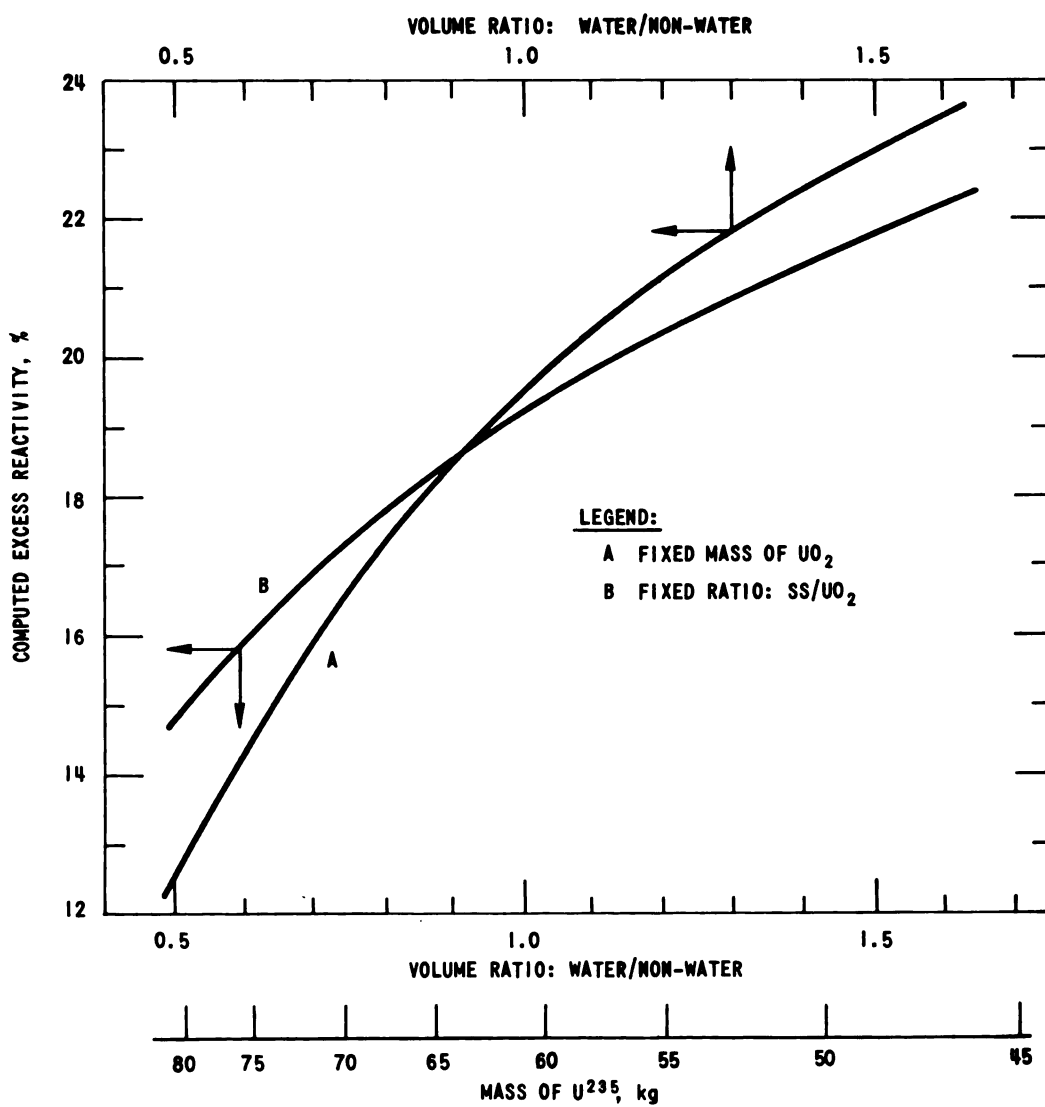


FIG. 13 AVAILABLE EXCESS REACTIVITY FOR CLEAN COLD AARR, WITHOUT GRADING OF FUEL LOADING, VS WATER CONTENT OF REACTOR FUEL ZONE

The effects of varying the fuel loading, and of varying the thickness of the ungraded portion of the reactor core, are of particular relevance to the discussion of the capability and limitations of the AARR, and these results are reviewed in detail in Section IV-F, following.

F. Reactivity and Neutron Fluxes as Functions of the Grading and the Thickness of the Fuel-Zone Annulus

Once the reactor fuel zone is made fairly heavily absorbing, additional fuel (U^{235}) has little effect on the peak thermal neutron flux in the ITC or in the beryllium, at a fixed power level. This point is illustrated in the curves of the radial distribution of U^{235} -fission activity reported for Criticality-Facility measurements on a wide range of fuel loadings.⁽²⁾ A variation in fuel loading from ~ 16 kg U^{235} to ~ 62 kg U^{235} reduced the peak flux (per unit power) in the ITC by only $\sim 10\%$. The average thermal neutron flux in the 100%- H_2O ITC was lowered by $\sim 18\%$, over this range of loadings.

The thickness of the fuel zone has a greater effect on the values of peak flux per unit power in the experimental facilities. This effect really is a composite of three principal effects. When the fuel zone thickness is decreased, but the total reactor power is held constant: (1) the power density near the inner and outer flux traps is increased; (2) the mean optical distance between the source and the receptor is decreased; and (3) the probability of leakage of high-energy source neutrons is increased.

The importance of an increase in the local power density over significant regions at the edges of the core has been studied in a more general way. Figures 14 and 15 show the relative contributions, to thermal neutron fluxes, from fission spectrum sources at various radial locations. Each is exponential, or nearly so; for example, the function $\exp(-0.18 r)$ is an excellent fit to the curves in Figure 14. Note that in each figure the shapes of the curves for group No. 15 and group No. 16 are almost identical. However, it should not be thought that additional flattening of the gross radial power distribution in the reactor core would result in a significant decrease in the peak flux per

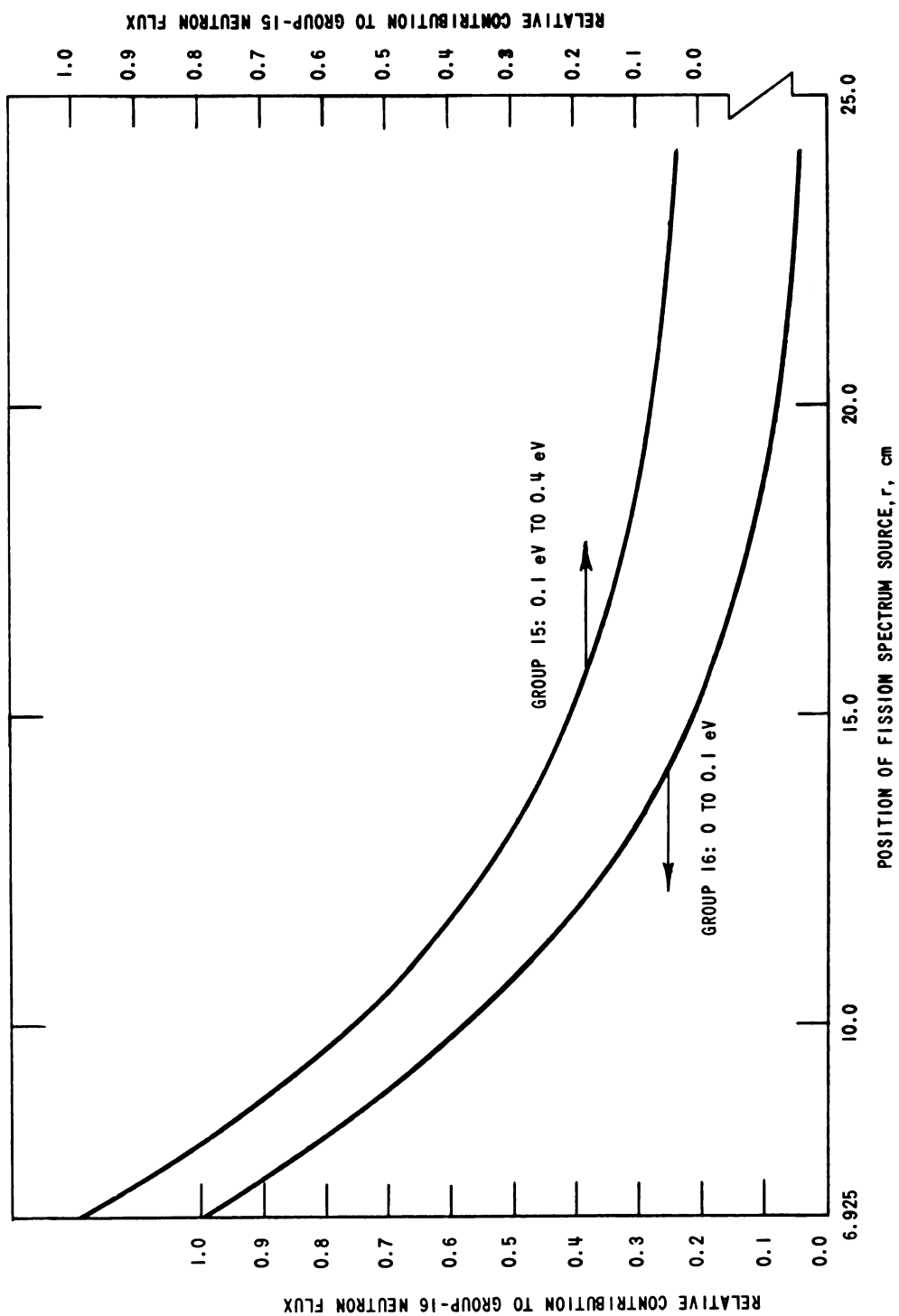


FIG. 14 RELATIVE CONTRIBUTION BY FISSION SPECTRUM SOURCES IN THE CORE TO THE THERMAL NEUTRON FLUX AT THE CENTER OF THE AAR INTERNAL THERMAL COLUMN (NORMALIZED TO UNIT CONTRIBUTION BY SOURCE AT INNER BOUNDARY OF REACTOR CORE)

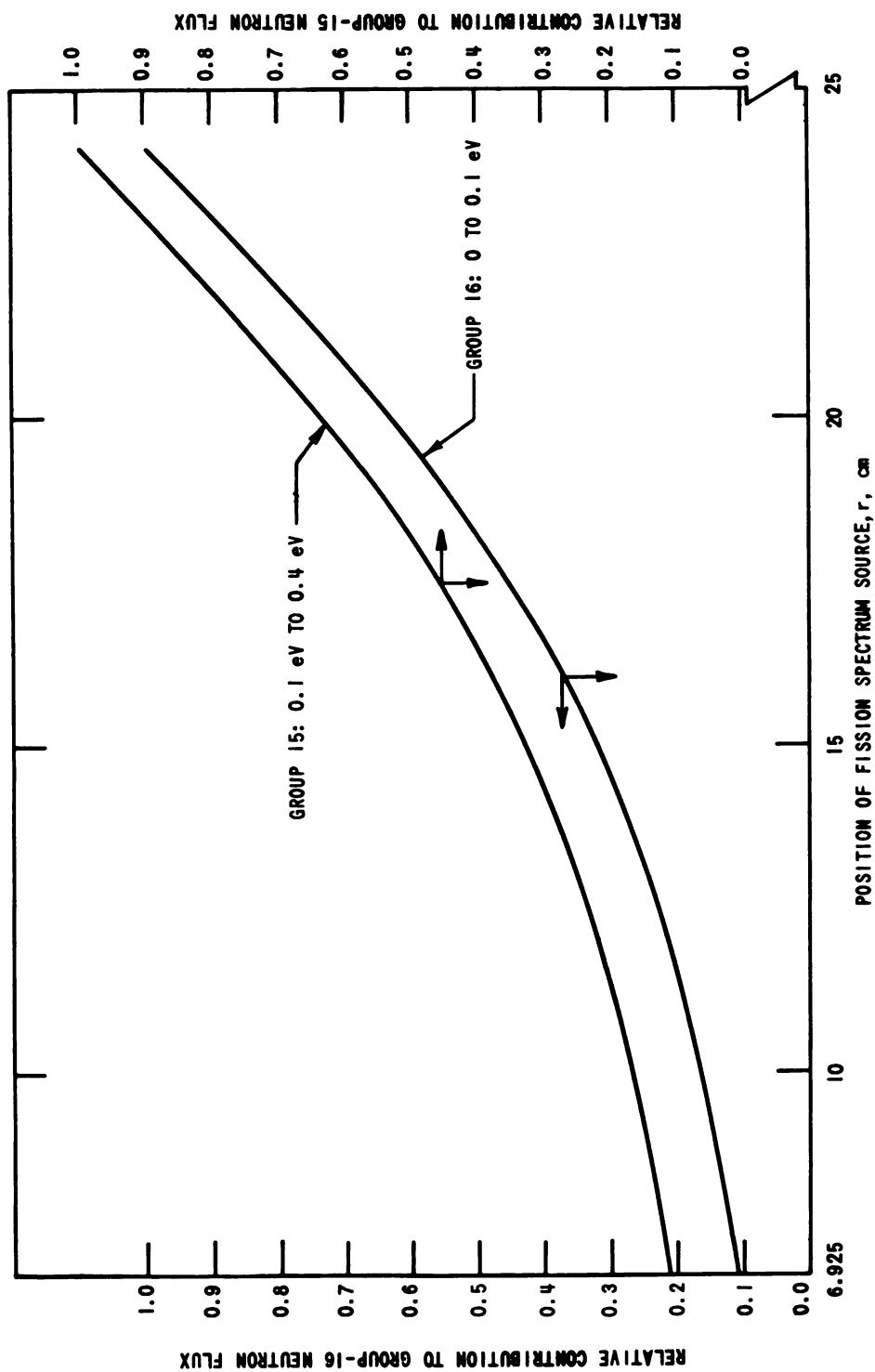


FIG. 15 RELATIVE CONTRIBUTION BY FISSION SPECTRUM SOURCES IN THE CORE TO THE PEAK THERMAL NEUTRON FLUX IN THE BERYLLIUM REFLECTOR (NORMALIZED TO UNIT CONTRIBUTION BY SOURCE AT OUTER BOUNDARY OF REACTOR CORE)

unit power in the experimental facilities. Computations have shown that, because the power density is peaked at both radial boundaries, a reduction in the edge powers and an accompanying increase in the power density in the central portion of the core offer such cancellations that the peak thermal neutron fluxes in the ITC and in the beryllium are changed by only a few percent. However, the additional grading of fuel which would be required to achieve this additional flattening of power density would reduce the excess reactivity available for core life. Not only would the reactivity of the fresh core be lowered, but the average reactivity loss per 1000 MWd of operation would be increased. This double penalty of the additional power flattening makes it undesirable to reduce the power peaking in the fuel zone significantly, in the AARR reference core.

Returning now to the effect on the peak thermal neutron fluxes outside the fuel zone when the fuel-zone thickness is changed, note that this means a corresponding change in the average power density at the edges of the reactor core, for fixed total power. If the core thickness is reduced, the increase in power density contributes to a higher peak thermal neutron flux, both in the ITC and in the beryllium. This effect is displayed in Fig. 16 for the design reference AARR, and, for comparison, for a much lighter (uniform) fuel loading. Only one curve is shown for the variation of the peak flux in the ITC since this curve for the reference loading is scarcely distinguishable from the corresponding curve for a fuel loading only $\sim \frac{1}{4}$ as large. Note also that the percentage change in the peak flux in the beryllium is almost the same as the percentage change in the peak flux in the ITC, over the entire range of thicknesses.

Figure 16 also shows the change in reactivity resulting from the change in core thickness. Note that there is little difference in the curves for the two different fuel loadings. Note also that each centimeter reduction in core thickness would cause a loss of $\sim 1\frac{1}{2}\%$ in reactivity. Therefore, a reduction in the reactor core thickness would mean that at fixed power the peak experimenters' fluxes would be increased but at a substantial cost in increased peak power density and in reactivity. For example, a 4-cm reduction in core thickness was found to correspond to a one-third reduction in core volume, a one-half increase in the average power density, a 40% increase in the peak power density, and a 35% increase in the peak thermal neutron flux in the ITC — at

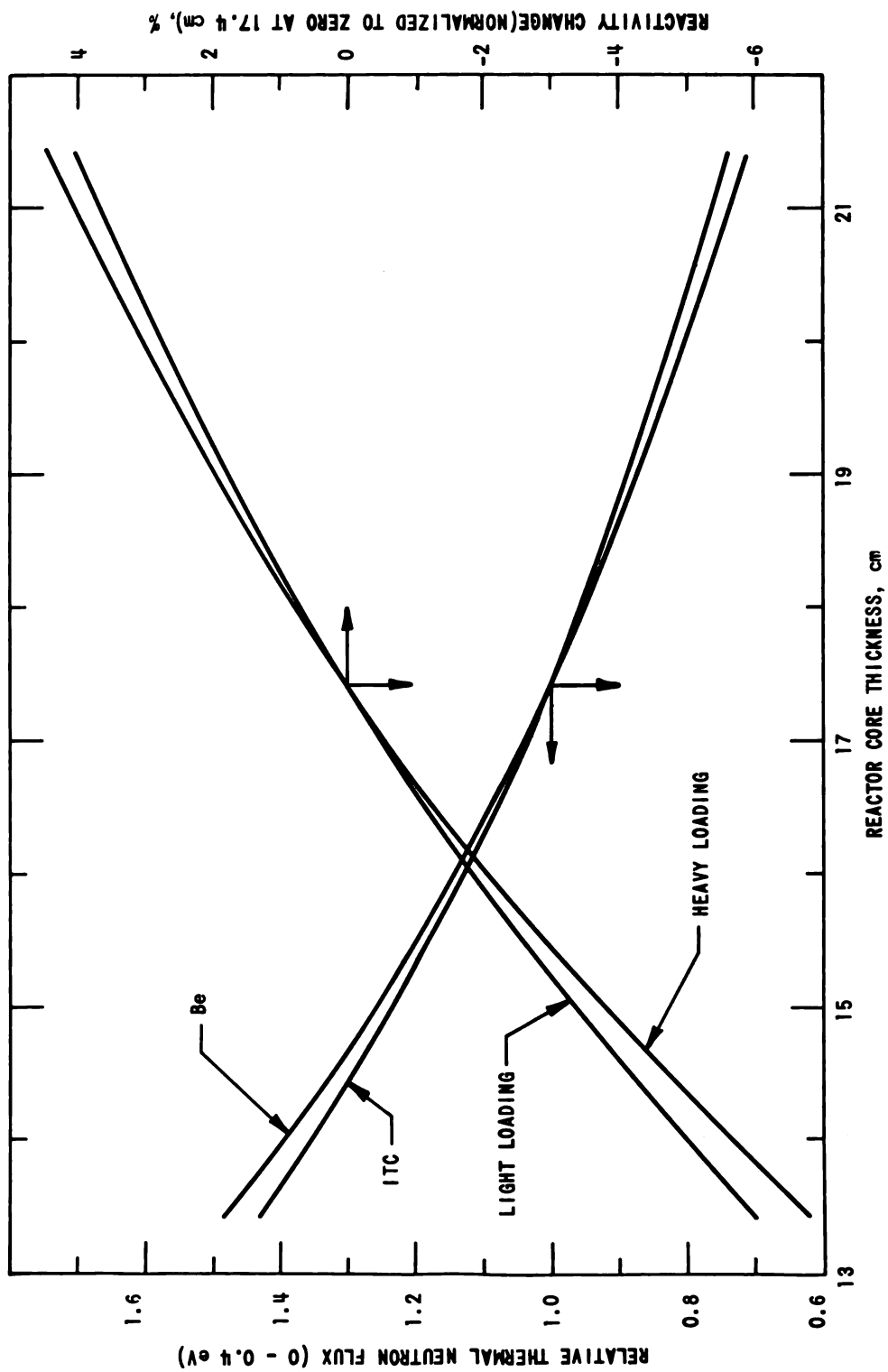


FIG. 16 EFFECT OF CORE THICKNESS ON REACTIVITY AND THERMAL NEUTRON FLUX

an initial cost of 6.7% in reactivity, which would be compounded by an increased loss in reactivity per 1000 MWd of operation.

The conclusion is that the AARR is essentially optimized, relative to radial power peaking and fuel-zone thickness, with the constraint of a long core life.

ACKNOWLEDGMENTS

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A U.K. STUDY FOR A HIGH FLUX BEAM REACTOR

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SUMMARY: There has been a growing interest in the United Kingdom in the use of neutron beams for studying solids and liquids on an atomic scale. To satisfy this requirement a design study has been undertaken of a reactor for producing intense neutron beams and incorporating many of the features required by potential users. The proposed reactor comprises a compact highly enriched core, cooled and reflected by heavy water. By means of a pressure tube assembly the core and coolant is at about 34 atmospheres and separated from the reflector, which is at near atmospheric pressure. The control and safety system is located immediately outside the core periphery and consists of 18 vertically operating control rods. With such an under-moderated core the thermalisation of the fast neutrons in the reflector causes the thermal flux to be a maximum just outside the core. At 100 MW this peak flux would be $\sim 2 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$. The present scheme provides for 5 major horizontal beam tubes up to 15 cm in diameter and 12 inclined and vertical holes. Special provision has been made for accommodating a high temperature ($\sim 1500^\circ \text{K}$) and a low temperature ($\sim 20^\circ \text{K}$) moderator, each viewed by 4 beam tubes, to increase the neutron beam intensities at wavelengths remote from that of the majority of reflector neutrons. A few small vertical irradiation positions are also included in the core and reflector to make use of the very high fast and thermal fluxes available.

The whole reactor is housed in a controlled-leakage containment building 37 metres in diameter and 24 metres high, large enough to allow substantial freedom for installing out-of-pile equipment.

1. INTRODUCTION

During the last few years the number of scientists working in the United Kingdom with neutron beams has substantially increased, and in particular many University Departments have become interested in the potentialities of neutron scattering for studying solids and liquids on an atomic scale.

With the increasing interest in neutron beams have come requests for much higher beam intensities, mainly associated with research in the solid and liquid states of matter. Three points have emerged, which are common to the application of high intensity neutron beams in these fields:

- (a) the crucial nature of some experiments which can be carried out using neutrons,

- (b) the number of such experiments which are hindered or which cannot be attempted because of lack of flux,
- (c) the increasing areas of useful applications of neutron beams (e.g. chemistry and biology) which higher fluxes would make possible.

To satisfy this growing need for more intense neutron beams a design study has been undertaken of a reactor incorporating many of the features required by potential users. A number of reactor designs, including pulsed devices, have been studied, and the latest efforts have centred around a reactor which, at a power of not more than 100 MW would have a peak thermal flux of $\sim 2 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$ in the reflector and accessible to neutron beam tubes. The reactor design has naturally led to a core operating at a high power density, which yields very high fast neutron fluxes. It has therefore been considered prudent to include a few in-core irradiation positions. Likewise a few are proposed for the reflector to make use of the very high thermal fluxes. However, the emphasis of the reactor design has been on the provision of high intensity neutron beams.

2. REACTOR DESIGNS CONSIDERED

With the aim of achieving a peak thermal flux of $\sim 2 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$ at a power of 100 MW, a number of possible systems were studied and compared with a reference design similar to the Brookhaven H.F.B.R. The reference design comprised a compact, highly enriched, uranium/aluminium-alloy plate-type core, cooled, moderated, and reflected by heavy water, and gave a peak flux of $2.6 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$ at a power of 100 MW. Similar fuel element dimensions (0.1 cm thick plates and 0.125 cm coolant gaps) were used for all the studies, and the power was fixed at 100 MW, with a heat flux limited to 500 watts/cm².

Calculations performed for a beryllium reflected, light-water cooled, core (of U/Al plates) gave a peak thermal flux of $1.9 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$. For a stainless steel clad fuel element core, with the required core loading of 43 kg of U²³⁵ and heavy water as coolant and reflector, the peak reflector flux was $1.5 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$. The maximum-to-average ratio of power density for this core was very high (about 7:1), and the spectrum was very poorly thermalised.

Though these cases were not completely optimised it was decided that it would be more profitable to concentrate on heavy water as reflector and a compact core with uranium/aluminium-alloy as fuel, with the alternative of light water or heavy water, at a pressure of about 34 atmospheres, as the coolant. Three types of system were studied:-

- (1) A pressure vessel design, similar to the Brookhaven H.F.B.R., with no separation between the coolant and reflector. This was the reference design.
- (2) A pressure barrel design in which the core was contained within one large pressure tube, enabling the coolant and moderator to be separated.

- (3) A pressure tube design in which each fuel element was in its own pressure tube, and so the coolant and moderator could be separated.

Two pressure barrel designs were studied, both using heavy water as a coolant. In the first scheme aluminium alloy was used for the pressure barrel. The thickness required, due to the pressure and thermal stresses, caused a severe reactivity penalty of 10% $\delta k/k$ and reduced the peak thermal flux by about 25% compared with the reference design ((1) above). In the second scheme a zirconium-niobium pressure barrel was studied. This material with its greater strength and smaller neutron absorption cross-section caused only a 1.7% reactivity penalty, and the peak thermal flux was reduced by only 5% compared with the reference design.

Serious consideration has been given to the second design. Its main disadvantage lies in the radiation damage to the zirconium-niobium alloy caused by the high fast flux. At present little is known of the effects of the fast neutron damage, particularly at low temperature, and it is considered that frequent changes of the pressure barrel might be necessary, causing possible inefficient use of the reactor. It was decided that if the neutron flux penalty was not too severe an aluminium alloy pressure tube was preferable to a zirconium-niobium pressure barrel, though there appears to be no data for either material of the effect of such massive fast neutron doses, as would be experienced during the life-time of the reactor. However, bearing in mind the existing data on these materials, and experiences to date with them, it is considered that aluminium alloy is the preferable material.

The pressure tube design was studied in some detail, both light-water and heavy-water cooled cores being considered. With heavy water as a coolant, the peak neutron flux was about 95% of that for the pressure vessel reference design, whilst with light water as a coolant, the corresponding figure was 70%. The reduced peak flux in the reflector, or the increased power required when using light water as a coolant, must be balanced against the ease of using light water and the savings on capital cost of the reactor. The decision is not necessarily clear cut. At Harwell, with its long experience in the use of heavy water, and with the possible difficulties associated with maintaining the separation of the light water from the heavy water, an all heavy-water system has been preferred.

Considerable work has been done on both pressure vessel and pressure tube systems. The pressure tube design, though giving a slightly lower peak thermal flux, has been preferred mainly because the reflector can be at near atmospheric pressure. This eases some of the stress problems which could arise on the complex neutron beam tube assemblies, and also allows the neutron beam tubes to be more easily replaced if the silicon build-up in the aluminium, due to the $\text{Al}^{27}(n, \gamma)$ thermal neutron reaction, became a problem.

3. DESCRIPTION OF PRESENT REACTOR DESIGN

3.1 The Core

Figures 1 and 2 show the main design details of the reactor. The core comprises 37 fuel elements on a triangular lattice to form a hexagon as shown. The six 'corner' elements are reduced in size to give a smooth contour to the periphery of the core. Concentric type fuel elements, as already used in the PLUTO materials testing reactor at A.E.R.E., are proposed. Each element, except for the six corner ones, contains 10 rings of fuel and has an outer diameter of about 6.5 cm and an inner diameter of 1.7 cm. The thickness of each concentric tube is 0.125 cm, and of each annular cooling gap is 0.125 cm. Experience with the present concentric elements indicates that adequate tolerances can be obtained. The core is approximately 75 cm high by 46 cm in diameter.

To accommodate the fuel elements and to provide individual pressure tubes for them, a large block of aluminium is bored out. The mass of aluminium alloy is reduced to a minimum by drilling small holes between the fuel element holes and by machining away the external surface of the block to follow the contours of the outer fuel elements. Above and below the core, the forging is shaped so that it can be directly welded to cylindrical piping, which is bolted to the main coolant circuit to facilitate replacement of this core assembly.

3.2 Control System

The nine main vertically operating control and shutdown absorbers are positioned outside the core, where the statistical weight reaches a maximum. When fully in, their lower tips are level with the bottom of the core. They are cylindrical, approximately 9 cm in diameter, and are equally spaced around the core, allowing enough space for easy removal from the reactor. These upper absorbers, besides acting as control rods, can be magnetically released to act as shutdown absorbers. They are balanced by nine lower 'auxilliary' absorbers, which can only be motored and operate below the core. When fully in, their upper tips are level with the bottom of the core. The two sets of absorbers should maintain a reasonably constant flux at the midplane of the reactor. Plate type absorbers were considered, but it proved difficult to design a control system in which they could be readily changed. Cadmium is proposed as the absorber, but because of its short lifetime in the reactor owing to burn-up, dysprosium or europium is also being considered. The heat generation in the control system due to neutron ($n\gamma$) and gamma ray heating is removed by circulating cooling water through the centre of the rods.

3.3 Reflector and Beam Tubes

The reflector is separate from the coolant and is contained in an

aluminium alloy tank 250 cm in diameter, which is pierced by a number of beam tubes disposed as shown in Fig. 3.

There are 5 horizontal tangential and one radial beam tubes all viewing the peak flux of the reactor, a $2 \times 10^{15} \text{ n cm}^{-2}\text{s}^{-1}$. At a greater distance from the core are two special facilities, designed to increase the neutron beam intensities at wavelengths remote from that of the majority of reflector neutrons. One facility will contain a high temperature moderator, viewed by 4 beam tubes, the other a low temperature moderator, similarly viewed by 4 beam tubes. It is proposed that the fifth tube, the upward angled one, of each of these 'multipods' is used for loading the moderator. At an even greater distance from the core, but still in a region of fairly high flux, are located 12 angled (with respect to the horizontal) and vertical holes and one through hole. All the single beam tubes can be up to 15 cm in diameter and are removable from the reflector tank, enabling different diameters and lengths to be fitted to allow for the adjustment of spectrum and collimating angle to suit certain experiments. The 'multipod' assemblies are also removable, and thus also allow some latitude in the size of the beam tubes. The disposition and angle of the beam tubes cannot be varied, for they are aligned with permanent holes in the thermal and biological shields.

In Fig. 3 all the beam tubes are shown having their maximum diameter of 15 cm. On the actual reactor the beam tubes' diameters will vary from 5 cm to 15 cm, and it is possible that a small number of low flux beam tubes would be excluded until experience had been obtained with the important high flux beam holes.

3.4 Irradiation Facilities

A small number of irradiation positions are considered for the core and reflector, the final number used depending upon the experience gained in the use of the reactor and the reactivity absorbed by them. A typical core loading for fast neutron damage work would use four fuel elements holes (~ 2.5 cm diameter) plus one of about 5 cm diameter in the central element, a number of inner fuel tubes being removed in both cases. In the reflector one 2.5 cm diameter experiment hole is proposed for the peak thermal flux, with five 5 cm ones in a flux of $\sim 10^{15} \text{ n cm}^{-2}\text{s}^{-1}$. There will be also a number of holes for 'rabbits' in $\sim 5 \times 10^{14} \text{ n cm}^{-2}\text{s}^{-1}$ flux where the thermal/epithermal flux ratio is particularly high.

3.5 The Shield

Surrounding the reflector tank is an iron-water-lead thermal shield. This protects the main biological shield from the gamma and neutron flux leaking from the core and also provides secondary containment in the even of a leak in the reflector tank. The surrounding main biological shield is of high density

concrete 245 cm thick. The material chosen for the shield is a concrete containing limonite and steel punchings and having a density of about 4.3 g/cc. The shield is designed to reduce the background to 100 counts per minute with a cadmium covered standard BF₃ counter. Above the core a different arrangement of shielding is contemplated as shown in Fig. 1. The thermal shield directly above the core is removable to give access to the core and permit removal and replacement of the pressure tube assembly. Thus the core block could be replaced by one of different core geometry, if this was thought desirable. All the neutron beam tubes are provided with either rotating or vertically operating shutters which reduce the radiation to ~ 100 mR/h, low enough to permit users to adjust their equipment while the reactor is at power. The beam tubes will also be flooded with helium to reduce neutron scattering losses from the beam.

3.6 The Coolant Circuits

The heavy-water coolant at a pressure of 34 atmospheres flows down through the core at 12 metres per second, and to prevent the core being uncovered following a break in the heavy-water circuit the coolant outlet pipes from the reactor are well above the level of the core. The heavy water is circulated by centrifugal pumps through light-water cooled heat exchangers, the 100 MW of heat in the light water being finally rejected to atmosphere by means of forced draft cooling towers. The heat appearing in the reflector heavy water due to heat generation in the heavy water, beam tubes, and control system caused by fast neutrons, gamma rays, and beta rays (from neutron activation) is removed by a small reflector cooling circuit at near atmospheric pressure.

3.7 Fuel Element Handling

A fourteen-day cycle (12 days at power, 2 days shut down) is proposed for the reactor. Fuel elements will be changed during the shutdown period of two days. Calculations have shown that even after cooling for four hours after shutdown from 100 MW the 'after heat' is sufficient to melt a fuel element within 2-3 minutes after being removed from water. It has therefore been decided that irradiated fuel must remain continuously immersed in H₂O or D₂O for at least one month. Owing to the difficulty of transferring the fuel element from heavy water to light water without loss or degradation of the heavy water, the present design proposes the use of an intermediate heavy water filled storage block in the top reactor shield, fuel elements being transferred to it in a heavy-water filled transfer flask. This flask is itself cooled by natural convection through an air-cooled heat exchanger.

3.8 Reactor Building

The reactor is housed in a cylindrical containment building of

reinforced concrete 37 metres in diameter by 24 metres high designed to provide the maximum working space for experimenters. The building has three main floor levels: the basement containing the reactor plant room, other operational equipment, and experimental space for the users of the vertical and low angled holes; the main experimental floor around the reactor, 150 cm below the core centre, with removable floor sections, to accommodate large vertical counter arrays; and the operations floor level with the top of the reactor. To provide space for experimenters using the upper angled holes a removable 'mezzanine' floor is provided 3 metres above the main experimental floor. There are also some laboratories and offices at the reactor top level and above it.

The building is designed as a controlled leakage containment building, in which under normal conditions air is removed from the building at a rate sufficient to keep the building at a slightly sub-atmospheric pressure so that air leakage will normally be into the building. Under normal running conditions the air will be drawn from the building through filters and discharged to atmosphere via a chimney stack 60 metres high. In emergency conditions the building air would be re-circulated through a clean-up plant, which would include filters and charcoal beds. A certain amount of air, equal to the in-leakage, would be drawn from this circuit to the stack to retain the negative pressure in the building. The building is designed to withstand 45 cm water gauge internal pressure, and should this be exceeded relief valves, which vent into the stack will prevent the building being damaged due to overpressure.

4. REACTOR PHYSICS CALCULATIONS

4.1 Core Calculations

A four-group model has been most frequently used for the critical size calculations; the thermal neutron group covered energies below 0.625 eV, and three fast groups were used with Ombrellaro energy divisions (1), 10 MeV - 0.821 MeV, 0.821 MeV - 5.53 keV, and 5.53 keV - 0.625 eV. Group constants were calculated from SOFOCATE (2) and NUFT programs and were set into the neutron diffusion program GRAM in (r,z) geometry to generate fluxes and adjoints, together with the system excess reactivity, or critical size. The geometric model most frequently employed in the computer calculations is shown in Fig. 4.

Calculations have been done for core loadings varying from 7 kg to 13 kg of U^{235} , with core volumes varying from about 90 litres to 120 litres at a fixed power of 100 MW. Table 1 shows the variation of excess reactivity, peak thermal flux, and maximum power density with increasing core radius and fuel loading at a constant core height of 53.34 cm. It is seen that as the U^{235}

Table 1

Variation of Excess Reactivity, Thermal Flux and Maximum Power Density with Core Size and Fuel Loading at a Constant Core Height

Core loading (kg U ₂₃₅)	7.2	8.0	9.0	10.0	11.0	9.0	10.0	10.0	11.0	11.0	11.0
Core volume (litres)	87.6	87.6	87.6	87.6	87.6	109.8	97.3	109.8	107.1	120.4	120.4
Excess reactivity %	11.6	13.1	14.1	14.9	15.5	14.6	15.4	16.0	16.6	17.1	17.1
Max. thermal flux ($10^{-15} \text{ n cm}^2 \text{ s}^{-1}$)	2.39	2.31	2.24	2.18	2.14	2.03	2.10	1.96	1.94	1.81	1.81
Max. power density (MW/litre)	2.25	2.32	2.41	2.50	2.60	1.86	2.22	1.92	2.04	1.75	1.75

Table 2

Variation of Excess Reactivity, Thermal Flux and Maximum Power Density with Core Size and Fuel Loading at a Constant Core Radius

Core loading (kg U ₂₃₅)	7.2	9	9	9	10	10	10	10	11	11	11
Core volume (litres)	87.6	87.6	109.8	109.8	87.6	87.6	97.3	109.8	107.1	120.4	120.4
Excess reactivity %	11.6	14.1	15.6	15.6	14.9	14.9	15.92	16.9	17.5	18.4	18.4
Max. thermal flux ($10^{15} \text{ n cm}^2 \text{ s}^{-1}$)	2.39	2.24	2.09	2.09	2.18	2.18	2.13	2.03	2.00	1.90	1.90
Max. power density (MW/litre)	2.25	2.41	1.75	1.75	2.50	2.50	2.16	1.81	1.94	1.63	1.63

loading is increased, the core volume remaining constant, the thermal flux peak decreases slightly. A similar variation in the thermal flux occurs when the core radius is increased. Table 2 is similar to Table 1, but the core radius is held constant, and again the thermal flux peak is decreased as the core height and volume is increased. These calculations were all performed on the basis of a pressure vessel form of design.

A number of calculations have been performed for a pressure tube design, with aluminium simulating experiments in the five fuel positions mentioned above. Table 3 shows typical parameters for cores of 114 litres volume containing 12.9 kg U^{235} . There is an appreciable gain in peak thermal flux together with a decrease in the maximum power density as the core height is increased and its radius is decreased. The core in Table 3 of 22.22 cm radius is close to that which is being considered as a possible core size for operation at 100 MW with some irradiation experiments in core. It should provide a peak reflector flux in operation of close to $2 \times 10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$.

Calculations have also been done for annular cores, and some results are given in Table 4. These results show the gain in reactivity and the reduction in maximum power density achieved with such cores. It can also be seen that if the excess reactivity is fixed, an annular core gives a higher peak flux than the corresponding cylindrical core, and illustrates the advantages of such cores for neutron beam work. If the actual operating core contains no irradiation experiments it is proposed that the central fuel element, and possibly other inner fuel elements, be removed to give maximum reflector fluxes. These possibilities are being studied in more detail. However, calculations indicate that the smaller loaded annular cores do require a greater excess reactivity for an operating cycle than do the cylindrical cores.

4.2 Neutron Fluxes in Beam Tubes

For all reactors the neutron flux in the beam tubes has always proved difficult to calculate. In such a reactor as that proposed it was deemed important to endeavour to calculate the flux depression caused by a beam tube and, more important, that caused by the multipods. The three-dimensional Monte Carlo program GEM (3) was used for this. For a 10 cm diameter beam tube the results yielded a flux in the beam tube of 0.9 of that for the same position in the reflector with no beam tube. For a multipod consisting of four beam tubes the corresponding figure was 0.6. The possible inaccuracy of these results was estimated to be 10-15% and was large because of the time required for the running of the program on the IBM 7090.

4.3 Reactivity Controlled by Control System

The proposed control rod system, when fully inserted, can be accurately simulated by a ring of nine rods surrounding the core. Estimates of the

Table 3

Reactor Physics Parameters for Cores of Constant Volume and Fuel Loading

Core radius (cm)	26.05	23.91	22.22	20.84
Excess reactivity (%)	15.1	16.75	17.6	18.3
Max. thermal ($10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$)	2.12	2.17	2.20	2.22
Max. power density (MW/litre)	2.02	1.92	1.83	1.74

Table 4

Reactor Physics Parameters for some Annular Cores

Core Outer radius cm	Core Inner radius cm	Core height cm	Excess reactivity (%)	Max. thermal flux ($10^{15} \text{ n cm}^{-2} \text{ s}^{-1}$)	Max. power density (MW/litre)
26.05	4.28	54.63	16.2	2.06	1.94
	6.0	56.33	17.3	2.01	1.89
	8.0	58.9	18.8	1.94	1.81
22.22	4.0	73.29	18.9	2.12	1.74
	8.0	84.44	22.1	1.93	1.57
23.3	0	73	17.9	2.09	1.69
23.3	3.3	73	17.9	2.15	1.72

reactivity controlled by this system have been obtained by using a Monte Carlo program MOCUP (4), which can treat such a control rod system. A correction (12%) had to be applied to the results obtained because MOCUP will only deal with control absorbers the same length as the core. The results indicated that the eighteen 9 cm diameter control rods should absorb $\sim 35\%$ $\delta k/k$ when fully inserted. This is adequate for control and safety requirements.

4.4 Reactivity Requirements

No theoretical estimates of the reactivity controlled by the beam tubes have been made. From results obtained from the Brookhaven HFBR zero energy work it is estimated that the beam tubes will control about 4% reactivity.

An allowance of 2% reactivity has been made for irradiation experiments and their thimbles.

Estimates have been made of the reactivity variation during an operating cycle. These calculations are sensitive to the assumed flux distributions, particularly their shape near the edge of the core. The reactivity absorbed by the xenon is also sensitive to the effective cross-section calculated for this reactor poison. After 12 days operation at 100 MW it is calculated that the xenon, samarium, and other fission products, and the burn-up of U^{235} , account for 8.5% reactivity loss.

The temperature coefficient of reactivity has been assumed to be approximately the same as that for the Brookhaven H.F.B.R., giving a total reactivity loss of 1.35%.

Assuming that 1-2% excess reactivity must be available at the end of the cycle, the total excess reactivity required is 17-18% $\delta k/k$.

4.5 Fuel Element Zoning

Fuel zoning has been proposed to reduce the maximum-to-average power density ratio, and calculations have been performed with a two-zone core to estimate the effect of such zoning on the peak reflector flux. The outer zone consisted of the outer ring of 18 elements, and the inner zone the remaining 19 elements. For the series of calculations, the U^{235} content in each zone was allowed to vary until the peak power density in each zone became approximately equal. The CRAM diffusion program was used for these calculations. To obtain the same excess reactivity with the reference core the U^{235} loading had to be increased 35% in the inner zone and decreased 17% in the outer zone to obtain the same maximum power densities in each zone. The maximum power density was decreased by 14%, and the peak thermal flux in the reflector changed negligibly ($\sim \frac{1}{2}\%$). In these calculations it was assumed that the neutron spectra in each zone did not change with the changes in the U^{235} loading. This can lead to errors, and a more detailed assessment is being performed.

A further reduction in the maximum power density can be obtained by

tapering the fuel axially near the top and bottom of the central elements. This is already being successfully done with special fuel elements for the materials testing reactors DIDO and PLUTO.

5. HEAT REMOVAL FROM THE CORE

The peak thermal flux in the reflector increases with increasing average power density and so it is desirable to design the core and heat removal circuit around this requirement. The average power density that can be permitted increases with increasing water velocity, increasing heat transfer area per unit volume (narrower coolant channels), and increasing pressure. It was judged that the average coolant gap, bearing in mind manufacturing tolerances, could not be less than 0.050 inches (1.25 mm) and that 34 atmospheres appeared to be a reasonable pressure limit when balancing the small increase in saturation temperature above this pressure against the increased cost, complexity, and core volume (increased ligament thickness of core pressure tube assembly). The water velocity was fixed by the considerations of experience, pressure drop through the core, and fuel element strength. These details fix the majority of the heat transfer characteristics of the core.

The fuel plate temperatures have been calculated from the well-known Seider and Tate equation using a coefficient of 0.024 instead of 0.027. This has been suggested by Gambill and Bundy of Oak Ridge.

For the calculation of sub-cooled burn-out power the correlation of Zenkevich and Subbotin (5) was used with an uncertainty factor of 1.3.

Parallel channel instability has also been considered, and the following criterion has been applied for the onset of this instability.

$$\frac{T_{\text{out}} - T_{\text{in}}}{T_{\text{sat}} - T_{\text{in}}} = C$$

where C is a constant between 0.75 and 0.8 which varies slowly with the hydraulic characteristics of the coolant channel.

T_{out} = temperature of water at outlet from hot channel

T_{in} = temperature of water at inlet of channel

T_{sat} = saturation temperature at outlet of hot channel.

This correlation has been obtained by Forgan (6) at Harwell from experiments on low pressure systems and is tentatively being used for the present studies. Better information spanning the desired pressure range is necessary.

The limitation on the maximum power density of the core is not imposed by burn-out, but by the build-up of an alumina film at the hot spot. Griess (7) at Oak Ridge has shown that when the fuel cladding temperature exceeds 177°C the build-up of an alumina film on the cladding becomes appreciable. The low thermal conductivity of the alumina, together with the high heat flux, produces an appreciable temperature difference across the film. Also if the alumina does form it has been shown that the aluminium fuel plate can suffer intergranular corrosion if water

seeps through a crack in the alumina film and comes into contact with the plate where temperature is above the saturation temperature. The criterion has been to specify a maximum metal temperature beneath the alumina lower than the saturation temperature.

The data for a typical core is shown in Table 5.

Table 5
Heat Transfer Data

Inlet water temperature	49°C
Inlet pressure	34 atmospheres
Coolant velocity	1220 cm/s
No. of fuel tubes per element	10
Max/mean power ratio	2.0
Hot channel factor	1.3
Hot spot factor	1.3
Coolant gap	0.125 cm
Hot spot heat flux	396 watts/cm ²
Hot channel outlet temp.	131°C
Alumina build-up	0.0034 cm
Alumina surface temp.	179°C
Clad/alumina interface temp.	232°C
Saturation temp. at outlet	233°C
Parallel channel instability margin	1.9
Sub-cooled burnout margin	2.1

It is seen that the temperature drop across the alumina at the hot spot is approximately 50°C. If this could be neglected parallel channel instability or sub-cooled burn out would limit the reactor power. The heat transfer area of the core could then probably be substantially reduced as could its volume, giving increased peak thermal fluxes. As the alumina can have such an important affect on the core design it is undoubtedly desirable that more work is performed on the rate of alumina build-up, and the effect of boiling at the alumina/clad interface.

6. GENERAL REMARKS

The proposed reactor is being designed around the requirements of the neutron beam users, and many features reflect these requirements.

Some features of the thermal shield and main shield designs which also affect other aspects of the reactor design could be simplified if longer beam paths, from tips of beam tubes to the reactor face, were allowed. Though it is appreciated that many experiments cannot accept the wide collimation angles possible with

such a shield it is considered that users should have available the potentialities of wide angles and maximum beam intensities, and with instrumentation development such angles could be used.

Liquid hydrogen moderators have been used in the Harwell reactors for many years and liquid hydrogen is proposed for the cold-neutron moderator in one of the 'multipods'. For the other 'multipod' graphite or beryllia blocks, operating between 1000°K and 2000°K are proposed. Experiments are being performed on the reactor LIDO with both liquid hydrogen as a cold moderator, and graphite and beryllia as hot moderators, to obtain data, which should lead to optimum geometrical designs for these forms of moderators.

The efficiency of the use of the reactor is dependent not only on the fraction of the total time available that the reactor operates but also on the efficiency of the experiments using its beam holes. Much consideration is being given to the instrumentation for the experiments and also to methods which would allow the experiments to be easily changed and which would demand a minimum amount of time for lining-up operations. Neutron conducting tubes have also been studied, and it is thought that the proposed shield design could accommodate present requirements for these tubes. Neutron conducting tubes have great advantages, effectively in reducing backgrounds, and it is considered that more experimental work on their design and uses is required before their potentialities can be fully utilised.

A zero-energy mock-up of the proposed high flux beam reactor is desirable. Initially this assembly would be used to determine the operating characteristics and limitations of the main reactor design, e.g. optimum core dimensions and U^{235} loading, power distributions and reactivity control, effect of beam tubes, etc. Later it would be used to optimise the beam tube arrangements to ensure that the correct neutron intensity and spectrum was obtained in relation, for example, to the gamma ray background. The detailed design of the 'hot' and 'cold' sources is dependent upon accurate estimates of the total heat generation within them, and this could be measured in the assembly. The effect of irradiation experiments on beam intensity can only be effectively done with a zero energy assembly. Also, it is inevitable that developments will be proposed for the main reactor, and these are best studied in detail with a mock-up. For the U.K. materials testing reactors, all these types of measurements have been performed on DAPHNE for many years and this has proved invaluable in reducing the amount of low-power time required on the main reactors, besides enabling experiments to be done which would have proved difficult on the main reactor. With the increased power and fluxes of the proposed reactor these points are even more important.

The proposed High Flux Beam Reactor has been designed around the requirements of potential neutron beam users. It offers about twenty neutron beams, independently useable, with wide convergence angles, controlled moderator temp-

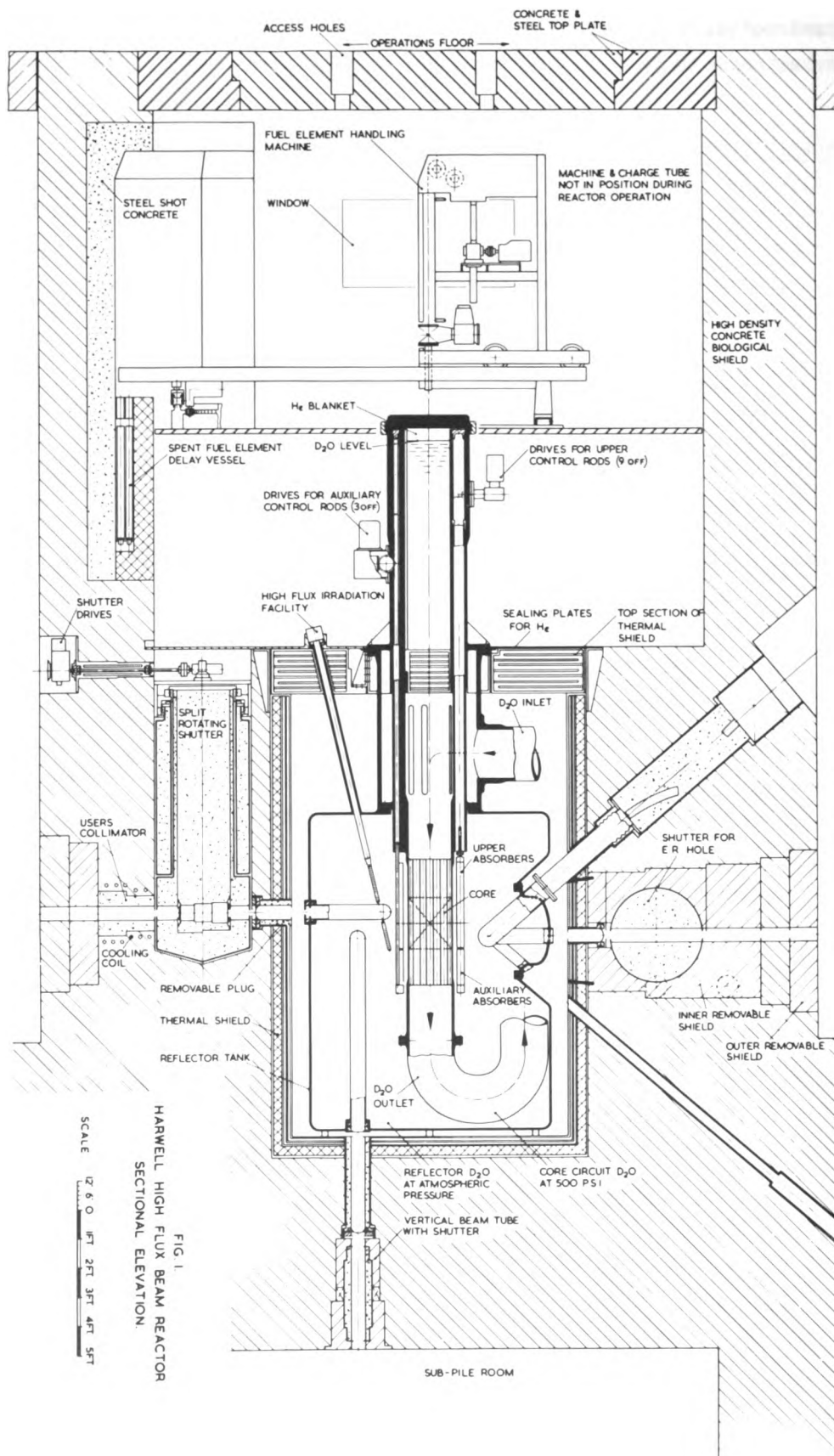
eratures and adequate space for instrumentation and should be a most useful tool for investigating a wide range of structural and dynamical effects in solids and liquids.

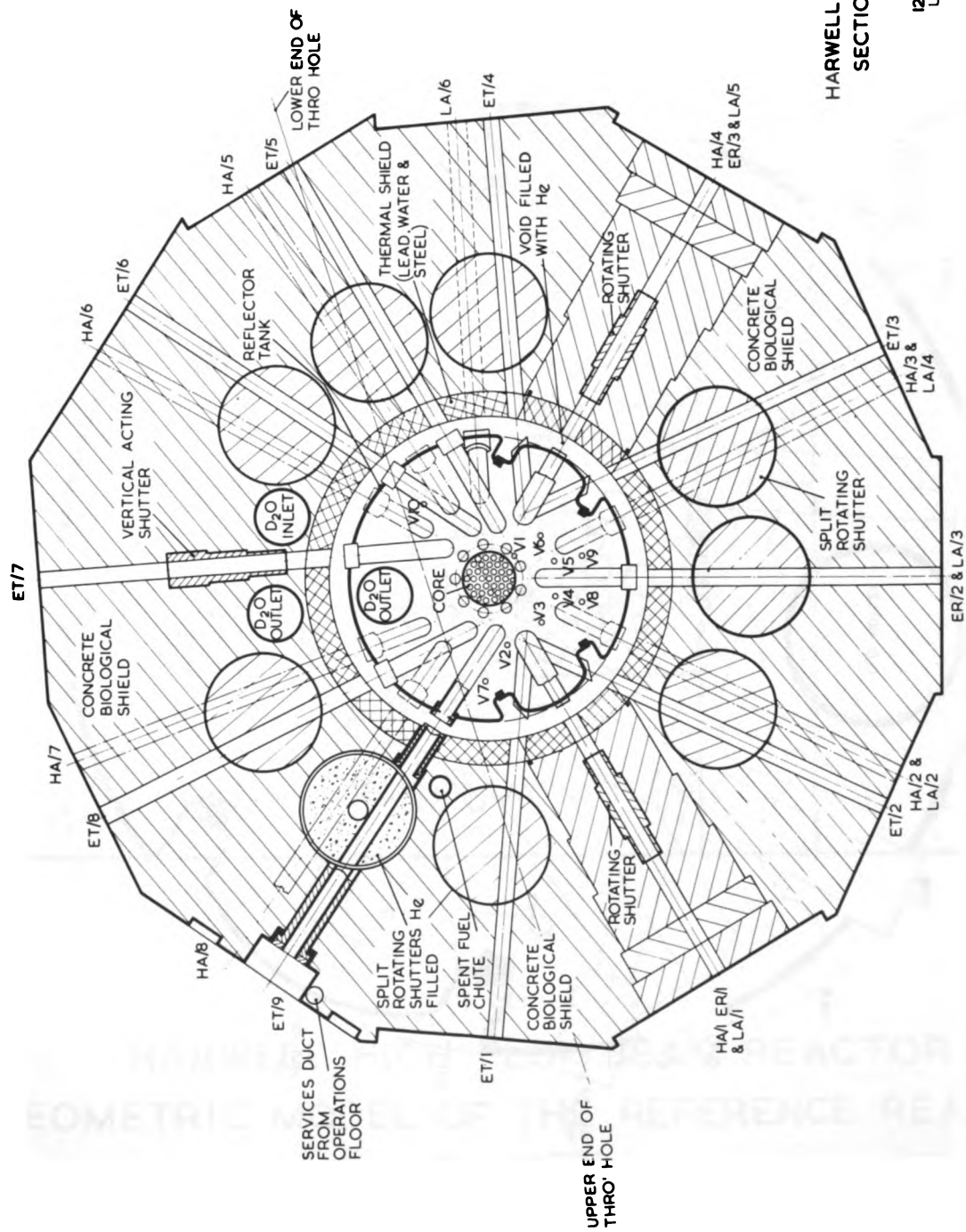
7. ACKNOWLEDGEMENTS

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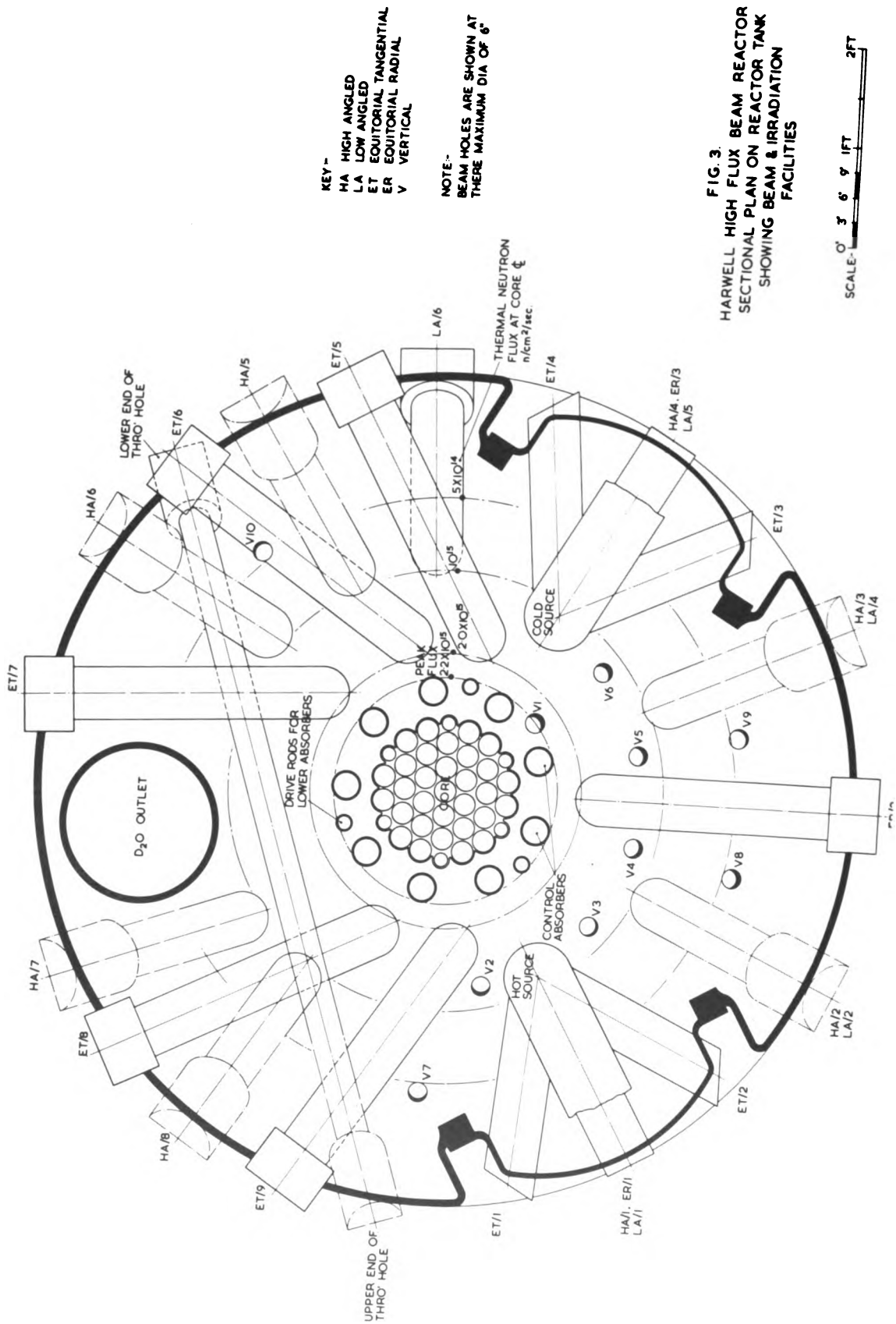


FIG. 3.
HARWELL HIGH FLUX BEAM REACTOR
SECTIONAL PLAN ON REACTOR TANK
SHOWING BEAM & IRRADIATION
FACILITIES

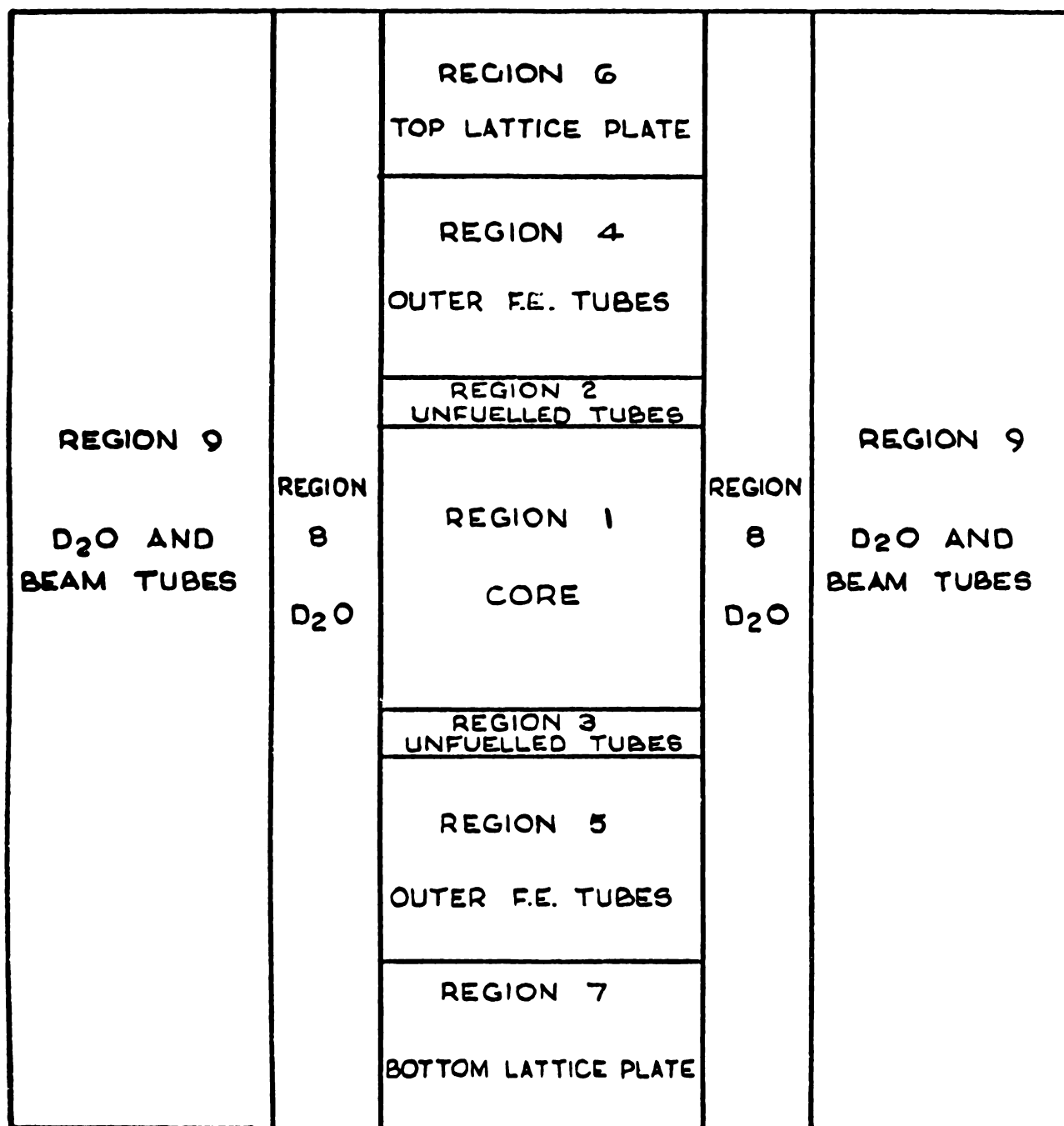


FIG. 4
HARWELL HIGH FLUX BEAM REACTOR
GEOMETRIC MODEL OF THE REFERENCE REACTOR
DESIGN.

PROJECT STUDIES FOR THE FRANCO-GERMAN HIGH FLUX REACTOR

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SUMMARY: As a main tool for the Franco-German Institute at Grenoble, a Very High Flux Reactor for beam tube experiments in nuclear, liquid and solid state physics will be built. The reactor will supply a (perturbed) thermal neutron flux of at least $1,0 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$ in a large heavy water reflector which is kept at a low pressure. A large number of beam tubes and other facilities, including a cold and a hot neutron source, are to be installed. The design aims to convenient operation of these facilities and to a maximum of flexibility. Removable experimental facilities and a partly removable shield as well as a pool have therefore been provided. The total reactor power is about 55 MW. A light water and a heavy-water cooled version of the reactor have been studied, each offering characteristic advantages.

1. INTRODUCTION

A Very High Flux Reactor for the production of neutron beams will be the central research tool of a newly formed joint franco-german research institute. This institute - to be called Max von Laue-Paul Langevin-Institute - will be located at Grenoble/Isère in close connection with the existing Centre d'Etudes Nucléaires de Grenoble. It will be sponsored by the French and German governments through the Commissariat à l'Energie Atomique and the Centre National de la Recherche Scientifique on one hand, the Gesellschaft für Kernforschung m. b. H. Karlsruhe on the other hand, both parties contributing equal parts to the investment and operating costs. The purpose of the institute will be to perform research in the fields of nuclear, solid and liquid state physics, taking advantage of the unique experimental possibilities offered by an advanced high flux reactor.

The institute's research programme will be carried out in very close collaboration with all interested research centers and universities within the two countries, close contacts with these groups being assured by the very form of the project organisation. Furthermore, the contracts will be laid out in such a way that other countries may join in at a later stage to participate in the utilisation of the High Flux Reactor.

While first talks between scientists on this project started at the time of the third Geneva conference, official negotiations were taken up in due course and have now been completed to such extent that the official contracts will be signed in the near future. Meanwhile, considerable progress has been made in the scientific and technical preparation of the project. Early in 1965, three joint working parties were formed, dealing with the following items:

- consideration of the experimental programme and its implication on the reactor design
- preparation of special experimental facilities, i.e. the cold and the hot neutron source
- consideration of the design of the reactor proper.

This paper deals essentially with the work of the latter party which has undertaken a number of detailed design studies during the last 15 months. Its work was greatly facilitated by earlier studies, performed by a joint Saclay-Grenoble group, and partly described in a report to the 1964 Geneva conference [1]. The working party consisted of scientists and engineers from the centers of Saclay, Grenoble and Karlsruhe, the latter being supported for part of the time by development contracts with Interatom G.m.b.H., Deutsche Babcock and Wilcox G.m.b.H. and with Nukem.

2. A REVIEW OF DESIGN PRINCIPLES

We may summarize the main objectives which guided our design work as follows:

- (a) The reactor shall supply a large number of high-quality intense thermal neutron beams. It will not have provisions for the large-scale irradiation of transuranium elements (i.e. no central flux-trap irradiation facility).
- (b) Though this is a high flux facility, the design does not aim at the highest flux available, but rather at the highest quality of

neutron beams and at optimum flexible experimental facilities. Therefore, the (perturbed) thermal flux at the beam tube tips is fixed to a value of $1.0 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$. As will be seen later on, this can be achieved with a total thermal reactor power of about 55 MW. It is true that higher powers and neutron fluxes can be achieved using existing or nearly available technologies, and indeed we shall see that even our design might be operated at higher power and flux. However, the possible gains do not appear to be very large and lead to increased fuel cycle costs.

(c) "High quality" of thermal neutron beams requires a very low background of fast and epithermal neutrons and γ -rays. A large number of such beams requests a large volume from which they can be extracted. As is well known and has been demonstrated through the Brookhaven HFBR [2], both these requirements can be met by a reactor concept where a compact, possibly undermoderated core is surrounded by a very large D_2O reflector from which the beams are extracted at a suitable core distance. We have accepted this scheme, however, modified insofar as we require the D_2O reflector to operate at a low pressure (≈ 2 atmospheres). This choice reduces the necessary thickness of the walls of experimental facilities and thus their heat rates and it allows the installation of novel kinds of experiments. It furthermore enables to design the experimental facilities in such a way that they can be removable and replacable. However, it introduces a major difficulty into the project since a shroud separating the high-pressure core region and the reflector, operating in a very high radiation field, is required.

(d) Very complex experimental facilities will have to be installed, e.g. a cold neutron source, a hot neutron source, a fission fragment mass separator with internal target, neutron conducting tubes for various energy ranges, a conversion electron spectrometer with internal target and several beam extraction tubes with very special shapes. These are considered to be integral parts of the reactor and their convenient installation, operation and maintenance as well as their implication on reactor safety have to be considered.

(e) In order to adapt the reactor to the developments in neutron physics, the experimental facilities must be capable of being completely altered at a frequency that has been tentatively chosen as three to four years. This requires a considerable amount of flexibility in the design of some reactor parts, especially reflector tank and shielding.

(f) As far as is possible, the reactor shall be based on existing technology. This holds especially for the fuel element material for which the well-established aluminium technology was chosen.

3. GENERAL DESCRIPTION OF THE REACTOR

The general layout of the reactor is shown in Figs. 1 and 2 (cf. also Fig. 6). The core is a hollow vertical cylinder, fueled with U^{235} , with an outer diameter of about 30 cm and a height between 80 and 110 cm. It is placed at the center of a spherical vessel of 250 cm diameter which contains the heavy water. This reactor assembly is located at the bottom of a pool enclosed at the sides by concrete shielding. Various horizontal and inclined beam tubes traverse the shielding and the water of the pool and penetrate into the reflector vessel. Control is achieved by a control element moving within the central hole of the core while shutdown occurs by a ring of absorber rods acting in the reflector close to the core surface. The cooling water is flowing downwards. Two basically different solutions as to the coolant medium have been investigated:

Solution I - H_2O cooling

Solution II - D_2O cooling

While many aspects of the reactor design are pertinent to both coolant versions, there are some characteristic differences which shall be outlined wherever necessary.

3.1. CORE

Studies were made for both pressure tube and pressure tank type cores, each with light or heavy water cooling. Although quite attractive for reasons of maintenance and handling, the pressure tube solution was rejected on neutronic reasons (poor ratio of maximum thermal neutron flux to total reactor power). The present design involves a fuel zone in the form of an annular cylinder with about 30 cm outer diameter. The height is 80 cm in the case of the H_2O -cooled and 110 cm in the case of the D_2O -cooled reactor. The internal diameters are 17 and 22 cm, respectively. The central hole contains a movable beryllium plug in the H_2O -cooled version or it is filled with D_2O , respectively.

The fuel zone is made up of aluminium clad parallel fuel plates, 1.27 mm in thickness. The width of the coolant channel is 1.8 mm. Mechanical assembly of the fuel plates might be done by the well-known ATR - or HFIR-

type techniques. Although more difficult to assemble, we are envisaging a modified single zone HFIR-type element as schematically indicated in Fig. 3. It is intended to manufacture this element in such a way that its outer aluminium casing can contain the internal pressure of the core (about 10 atmospheres) and can thus serve as a pressure shroud.

3.2. REFLECTOR VESSEL

The reflector vessel is a sphere, about 2.5 m in diameter, this size being set by the neutronics. The pressure is about 2 atmospheres. The whole vessel can be exchanged and aluminium is used mainly in order to keep its activation low in view of the handling. There are about 16 beam tube penetrations in the vessel.

3.3. PRESSURE SHROUD

In the heavy-water cooled version, the shroud must withstand the internal pressure (about 10 atmospheres), separate the regions of different flow rates, and avoid transfer of contamination between the core and reflector region after any fuel melting. No high degree of tightness against leaks into the reflector is necessary. If light water is used as a coolant, the core shroud must also be tight against leaks between core and reflector regions. The separation of the light water and the heavy water must be excellent. The radiation level at the core-reflector interface is very high (about $1.2 \cdot 10^{22}$ fast neutrons per cm^2 and per year). In view of neutron economy, low absorption materials, i.e. zirconium or aluminium alloys, must be used, whose behaviour under the irradiation considered is not completely known. For this reason, there are certain technical risks and one must be able to replace damaged components.

A possible solution for the H_2O -cooled version is schematically indicated in Fig. 4. The use of a fuel element whose outer aluminium casing retains the pressure and is thus the stressed wall is considered here. This casing is thus replaced with the core after an irradiation of about $1.4 \cdot 10^{21}$ fast neutrons/ cm^2 in a cycle. This dose is low and there is no special technical difficulty. Leakage around this wall passes into a water layer with a recovery circuit. A tight shroud (seal shroud) is still necessary between this water layer and the reflector but this wall is under low stress. If it will be made of Zircaloy, it will be 1.5 mm thick with a pressure stress of 0.375 kg/mm^2 . The axial stresses are limited by a bellows. The irradiation rate is approximately that of the primary pressure shroud. Since the stresses are much lower, the behaviour under

irradiation is less critical and it is expected that the seal shroud will operate safely for extended periods. It is nevertheless foreseen to exchange this seal shroud, and several methods of replacement have been investigated. - For the heavy-water-cooled core, it is sufficient to use the outer casing of the fuel element as a pressure barrel since leakage into the reflector can be admitted.

We have also studied designs which do not rely on the pressure retention by the outer fuel casing. In this case, an additional pressure shroud is used which need not be tight but must be easily replacable.

3.4. POOL

The reactor is at the bottom of a pool from 4.5 to 6 m in diameter. The pool depth is 12.6 m. This pool makes superfluous any need for a thermal shield which would reduce the flexibility of the experimental facilities. It also dispenses with the need for shielding above that would severely complicate the installation, maintenance and replacement of the complex facilities in the upper part of the reactor (cold source, hot source). The thickness of the lateral water layer between the reflector and the wall of the pool is 1 m, sufficient to yield a very low activation of this wall. It is thus possible to work in the pool after removal of the reactor vessel. The pool water is being circulated and there is a protection hot water layer at its surface in order to keep the radiation level at the surface within reasonable limits.

3.5. CONTROL SYSTEM

Control is achieved by a control element moving within the central cavity of the fuel region. The movement is in the vertical direction, the control rod drive mechanism being placed below the reactor. In the case of solution I (H_2O -cooling), the control element is a beryllium cylinder with a diameter of 15 cm which displaces the water. Replacing water in the central cavity by beryllium increases reactivity (due to the lower absorption and partly due to the $(n,2n)$ reaction on Be^9). The Be cylinder thus moves from below into the core during the cycle. Its total reactivity worth is 9.5 %. We do not expect radiation effects on the beryllium to represent a serious problem. In the case of solution II, the control element is a Nickel absorber tube, 14 cm in diameter and 0.7 cm in thickness. Its reactivity worth is about 11 % and it is moving out from the core during the cycle. In both solutions, the deformation of the flux in the reflector due to a control

element movement is very small. There are also no serious hot spot problems caused by flux deviations within the core since the element works in the region of core coolant exit so that the flux shift is towards the cold end of the core.

The safety rods have been placed in the reflector, close to the core-reflector interface. They move in the vertical direction and will also be operated from below the reactor. The driving force will thus have to be supplied by an external agent. About 25 % in reactivity have to be controlled by the safety system and it has been shown by detailed calculations that this can be achieved with 6 to 8 rods even in the case of the light water cooled reactor where the statistical weight for thermal neutrons at the core-reflector interface is lower than in the heavy-water cooled version.

3.6. REACTOR BUILDING

A schematic drawing of the reactor building is given in Fig. 5. The reactor is located strongly excentric in the experimental area. This will permit the use of neutron flight paths up to 30 meters long for some experiments, with these flight paths entirely inside the building. The floor of the experimental area is essentially the ground level in order to conveniently perform experiments with neutron guides which extend to outside the building. A basement of about 30 meter diameter is foreseen which will partly be used for reactor equipment (control and safety rod drives), partly for experiments (e.g. experiments with inclined beam tubes). A floor separates the experimental area from an area above which will be used for operations and vertically oriented experimental facilities. This floor prevents, with appropriate ventilation equipment, any spreading to the experimental area of accidental contamination of the operation area. The experimental area like the entire reactor building will be so built that no radioactive material can escape to the environment, under general conditions of safety norms. The detailed requirements as to tightness and possible pressure buildup are still under consideration.

An adjacent auxiliary building is planned. This will house various auxiliary components like those for water cleanup, reflector circuits components, equipment for control of effluents, ventilation and filtration units, a workshop, emergency diesel generators, facilities for the handling of large radioactive components in the pool, the hot cell, decontamination equipment, and probably the primary circuit pumps, valves and heat exchangers.

It is connected with the reactor by the transfer canal.

3.7. MAIN COOLING CIRCUITS

Some characteristic data for the main cooling circuits are given in table 1 and 2. As a secondary coolant, ground water is being used which is available at Grenoble in sufficient quantity and at a low temperature level ($< 13^{\circ}\text{C}$). The reflector cooling circuit is identical for both solutions. Apart from removing the heat in the reflector by core and capture γ -rays and by fast neutrons, the heavy water flow in the reflector must prevent boiling at the tips of the beam tubes and at surfaces of other components. The reflector has a high tritium activity (total inventory about $6 \cdot 10^5$ curies after 2 years operation). An intermediate light water circuit is used in order to exclude accidental contamination of the ground water in case of rupture of a heat exchanger.

Table 1 CORE COOLING

	H ₂ O cooling	D ₂ O cooling
Power to the coolant	53 MW	51 MW
Temperature of ground water	13 $^{\circ}$ C	13 $^{\circ}$ C
Temperature at core entry	30 $^{\circ}$ C	30 $^{\circ}$ C
Temperature at core exit	60 $^{\circ}$ C	60 $^{\circ}$ C
Normal primary flow rate	1,800 m ³ /h	1,800 m ³ /h

Table 2 REFLECTOR COOLING

Cooling required	4.5 MW
Ground water temperature	13 $^{\circ}$ C
Secondary water exit temperature	30 $^{\circ}$ C
Cold leg temperature, intermediate circuit	20 $^{\circ}$ C
Hot leg temperature, intermediate circuit	40 $^{\circ}$ C
Reflector water temperature at vessel exit	60 $^{\circ}$ C
Reflector pressure at vessel entry	2.2 bars
Nominal reflector flow rate	150 m ³ /h

The core coolant circuit in the case of solution I is straight forward, as common to many research reactors. A holdup tank will be used in order to reduce the N^{16} activity of the heat exchangers. Pumps and exchangers are located in the auxiliary building. No unusual requirements on leak tightness exist. Automatic valves in the pool will be used, which, in case of a pipe rupture, would allow natural circulation cooling of the core by the pool water ¹⁾.

The core coolant circuit for solution II is more complex. It contains several additional circuits (recombination, Helium circuit) and certain precautions must be taken in view of the use of heavy water (minimization of volume, minimization of leaks, control of tritium contamination, avoidance of tritium leakage into the secondary cooling water). Since the tritium inventory in the core is very much smaller than in the reflector, no intermediate cooling circuit will be used which would be very expensive. This makes very careful measures necessary in order to control leak-tightness of the heat exchangers. Because of the high coolant costs, efforts have been made to choose locations of exchangers, pumps and valves in such a way as to save on D_2O inventory. It seems possible to locate them within the pool or directly above it in the operation part of the building but the most convenient solution would be to install them in the auxiliary building (leading to a total D_2O investment in the core circuit of 17 m^3). In any case, the components will be placed above the core level so as to avoid any risk of coolant loss in the core region.

3.8. FUEL HANDLING

In solution I, loading and unloading of the core are performed entirely under water as with all pool reactors. The water provides shielding, natural circulation cooling and direct viewing of the operation. This is the main advantage of the light-water cooled version. For the D_2O -cooled

1) A study of flow reversal, which is needed here, will be done in the near future.

version, a special fuel handling system is required for unloading and transport of a burned fuel element as well as for loading a fresh one. We have investigated several solutions for this fuel handling system. The most convenient solution consists in a heavy-water cooled unloading machine which would assure shielding and cooling of the burned element during unloading. It could operate rapidly and safely and one could dispense with permanent equipment mounted above the pool. Unfortunately,

this is a quite expensive scheme. An alternative solution, illustrated in Fig. 6, consists in a shielded storage room in a rotating plug in the upper part of the pool. The spent fuel element is stored there for an adequate cooling time (several weeks), still included in the primary circuit. For later transfer of the element to the H_2O pool, a simple system can be used. The disadvantage of this method is that part of the pool surface is occupied by fixed equipment and thus the area above available for experimental equipment will be reduced. In a third variant, a small leak-tight shielded room is placed above the pool. After a reactor shut down of 24 hours, when the afterheat is about 0.19 % of the reactor power, the core is drawn from the heavy water into this shielded room with the removal of the afterheat assured by an air stream. A transport system then moves the core from its location above the reactor to above the pool and it is then lowered into the pool. This method, which is similar to the method successfully used at HFBR, again interferes considerably with the vertical experimental facilities.

The fact that the whole core has to be handled as a single piece does not pose very serious criticality problems.

4. PHYSICS CALCULATIONS

Extended neutronic and thermodynamic studies were carried out in order to arrive at an optimum core design. In these calculations, the flux perturbation effect by the beam tubes in the reflector which is difficult to treat exactly was taken into account by requesting a maximum thermal flux of $1.5 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$ in the reflector in the absence of beam tubes. The perturbation due to the beam tubes will be from 20 to 30 % and thus we can be sure to reach a real flux of at least $1.0 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$ at the tips of the beam tubes. At this flux level and within the limits set by the fuel, the core design was optimized as to yield the lowest fuel cycle costs.

4.1. NUCLEAR DATA AND COMPUTATIONAL METHODS

Most of the calculations were carried out in two-dimensional four or five group diffusion theory, using mainly the code ALCI but also TWENTY GRAND, PDQ and DYXY. The group energy boundaries were 10 MeV, 0.821 MeV, 5.5 keV and 0.625 eV.

For the calculation of the fast and epithermal group constants, the FORM version of the MUFT IV code was used. The input data were taken from the

present Saclay MUFT IV cross section library with the exception of some deuterium data which were taken from ~~EMERICH~~ [3]. No leakage was assumed ($B^2 = 0$) in these MUFT IV calculations and it was shown by additional calculations that the group constants thus obtained slightly overestimate the reactivity (by 200 pcm for solution I and by 750 pcm for solution II). This effect was corrected for.

Calculation of thermal group constants posed particular problems since the thermal neutron spectrum is very strongly space dependent, undergoing a strong hardening as one goes to the interior of the core. As a simple approximation, a two thermal group model was developed, one group representing the thermal neutrons born by thermalization within the fuel zone, the other representing those borne elsewhere [4]. These two groups diffuse independently and without mutual transitions. The constants for both groups are found by averaging microscopic data over the actual spectra within the two groups as calculated by the THERMOS code. This procedure leads to a correct neutron balance for each group and in particular to a good description of the spatial power distribution.

4.2. THERMODYNAMIC DATA

The operating pressure was chosen in such a way as to prevent continuous boiling under normal operating conditions and to assure a margin of 25 % against redistribution (secondary burnout). The coolant velocity was chosen to be 14 m/sec and the coolant inlet temperature as 30° C. The following principal experimental correlations, which were largely investigated by the Service de Transfert Thermique de Grenoble, were used:

Friction coefficient in the liquid phase: COLEBROOK-MOODY with a dimensionless surface roughness $\xi = 0.002$ for the case without heat transfer; modified according to MC ADDAMS [5] for the case with heat transfer.

Heat transfer to the liquid phase: COLBURN

Superheat in the presence of local boiling: FORSTER-GREIF [6]

Redistribution: OWENS-SCHROCK [7]

The usual formulae for the pressure drop at the core extremities were employed.

The various uncertainty factors are summarized in table 3. Uncertainties were calculated by multiplication, assuming that all unfavourable effects occur simultaneously.

Table 3 UNCERTAINTY FACTORS USED IN THERMAL CALCULATIONS

Ratio	<u>Friction Factor in hot channel</u> average friction factor	1.10
Correlation of COLBURN		0.85
Correlation of FORSTER-GREIF		1
Correlation of OWENS-SCHROCK		1.20
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Uranium loading per plate		1.02
Local uranium density over an area of 1 cm ²		1.15
Average value of width of hot channel		± 0.2 mm
Local width of hot channel		± 0.2 mm
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Power measurement		1.05
Flow measurement		0.98
Pressure measurement		0.98
Inlet temperature measurement		1° C

4.3. STATIC STUDIES FOR THE PROPOSED CORES

The basic composition and geometry of the cores investigated can be taken from Figs. 7 a and 7 b. The metal to water ratio in the fuel zone follows from thermodynamic considerations ¹⁾. The specific uranium²³⁵ loading is chosen to be 185 g/l. This is high according to the state of the art, corresponding to 33 % by weight of uranium in the meat. However, increasing the uranium loading is very worthwhile since it increases the flux to power ratio. Note that Nickel and Boron respectively are located at the end of the

1) It turned out that the specific power extractable passes through a maximum for a coolant channel width of 1.8 mm.
fuel plates in order to reduce the flux peaking.

Data calculated for the two cores are presented in table 4. The "rendement" as given in the first line is the ratio of the maximum thermal neutron flux in the reflector to the total reactor power ¹⁾. Rendement, total power and power densities were calculated for half inserted control elements while the excess reactivity given in the last line is for the control element in a position to yield maximum reactivity.

The amount of excess reactivity being fixed by fuel cycle considerations, the geometrical dimensions of the reactors (inner and outer radius, height) were chosen in such a way as to yield a maximum rendement and thus a minimum thermal power which is roughly proportional to the fuel cycle costs. This choice is influenced by a restraining condition of thermodynamics i.e. the maximum surface temperature shall not exceed $\approx 175^{\circ}\text{C}$ at the beginning of the cycle. It turns out that the reactor height has a very strong influence on the rendement, for instance it increases by 8 % when the height is increased from 65 to 80 cm in the H_2O -cooled reactor. Unfortunately, thermodynamics becomes more difficult at the same time since the cooling channel is made longer and the volume is decreased. The inner radius also influences the rendement strongly. At the same reactivity, the rendement for a D_2O -cooled core with central cavity is some 15 % higher than for a compact core. Use of the cavity also improves the power flattening; the ratio of peak-to average power densities is 55 % lower than in the compact core.

The surprising difference between the two solutions is the volume which is some 30 % higher in the D_2O -cooled core. This is because the D_2O -cooled core is almost completely externally moderated. Neutrons thermalized in the reflector can either escape to the outside or reenter the core to

1) Since most reactor codes achieve criticality by dividing $\sqrt{\lambda}$, the number of prompt neutrons generated in fission, through the eigenvalue λ , they yield an incorrect value of the rendement. The above rendements were calculated achieving criticality by homogenous poisoning and are thus the correct ones.

continue the chain reaction. In order to compete successfully with this external leakage, the core must be sufficiently large. In contrast to this, a considerable amount of neutrons in the H_2O -cooled reactor is moderated inside the core and the chain reaction is carried only to a small part by neutrons returning from the reflector. As a consequence of the larger volume, the power densities and temperature levels in the D_2O -cooled reactor are lower. In fact, the thermodynamic limit is not reached. - Note that the rendement of both solutions is very close. This is largely a consequence of the use of the beryllium insert in the H_2O reactor; using a compact core or an aluminium or water filling in the central cavity leads to less favorable results.

Table 4 RELATIVE CHARACTERISTICS OF FRESH CORES

	D ₂ O Core	H ₂ O Core
Initial Rendement (10^{13} neutrons/cm ² sec per MW)	2.75	2.65
Volume (liter)	46.6	34.76
Inner radius (cm)	11	8.5
Outer radius (cm)	16	14.5
Height (cm)	110	80
Initial power (MW)	54.55	56.6
Ratio $\frac{S_{\max}}{S}$ (peak - to average power density)	2.09	2.12
Effective maximum heat flux (W/cm ²)	407	577.5
Maximum Power Density (MW/l)	2.24	3.24
Average Power Density (MW/l)	1.07	1.53
Temperature at Hot Spot (°C)	140	177
Excess Reactivity (p cm)	23,150	21,550
Pressure at core inlet (bars) ¹⁾	10,5	9.7

1) There is an additional criterion on pressure. If during operation a break in the oxide film occurs and if we want to avoid formation of steam in this break, we have to chose a pressure such that the saturation temperature will be larger than the plate temperature. This criterion would impose 25,3 bar for the H₂O- and 12.7 bar for the D₂O-cooled reactor. Choosing these higher pressures would also increase the margin against redistribution.

No attempts have been made so far to flatten the power distribution in the core by fuel zoning or by use of heterogeneously distributed poisons in the radial direction. A considerable improvement in thermodynamic performance might be achieved in this way.

4.4. FUEL CYCLE STUDIES

Fuel cycle studies were made on the assumption that the maximum burnup achievable is $1.25 \cdot 10^{21}$ fissions/cm³, equivalent to a 50 % burnup for a loading density of 33 % U²³⁵ in the meat, at a fuel temperature not exceeding 230° C. This is well within the limits of irradiation experiments carried out on U₃O₈ and UAl₃ meat material [87]. 50 % maximum burnup corresponds to about 30 % average burnup. We have found that higher values of the average burnup may lead to higher fuel cycle costs since the requirements in initial excess reactivity become too high, leading

to a reduction of the rendement. It appears as if 50 % maximum burnup leads approximately to a minimum in fuel cycle costs.

A two-dimensional burnup study was made on the two above cores, using the TURBO multigroup code. Xenon, samarium and stable fission products (the latter with 61.6 barns/fission at 2200 m/sec in the thermal and with 20 barns/fission in the epithermal group) were separately taken into account. Since the rendement increases over the cycle, the reactor power was decreased in order to keep the maximum thermal flux constant at $1.5 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$. Some results of these calculations are summarized in table 5. It is seen that the fuel cycle length is greatly different for both reactors due to the differences in power density. The build-up of an oxide layer on the cladding material was calculated using the well-known GRIESS correlation [9], assuming a water pH of 5 and cladding corrosion rate equal to that of aluminium 6061-0. Due to the formation of this layer, the hot spot temperature increases at the beginning of the cycle, passes through a maximum and later on decreases due to the "natural" improvement of the power flattening. It is seen that the surface temperature stays within reasonable limits; in the D₂O-reactor, it is surprisingly low. The total reactivity loss includes the Xe and Sm effect (which itself is about 5000 pcm) as well as the effect of Uranium depletion and stable fission products. Homogenously distributed B¹⁰ has been used as a burnable poison in order to compensate for that part of the reactivity not bound by fixed absorbers nor by the control element.

Table 5 RESULTS OF TURBO CALCULATIONS

	D ₂ O core	H ₂ O core
Maximum burnup	50 %	50 %
Average burnup	25.5 %	27 %
Cycle length	38 days	26 days
Thermal Power at zero time	54.55 MW	56.6 MW
Thermal Power at end of cycle	52.25 MW	52.7 MW
S _{max} /S at zero time	2.09	2.12
S _{max} /S at end of cycle	1.51	1.5
Maximum surface temperature	150° C	210° C
Thickness of oxide layer	< 20 μ	38 μ
Total reactivity loss over cycle	14,820 pcm	14,140 pcm

Reactivity kept in burnable poison, zero time	7,250 pcm	9,315 pcm
Reactivity kept in burnable poison, end of cycle	970 pcm	1,863 pcm

In table 6, a reactivity balance for both reactors is given. The beam tube reactivity effect for the D_2O reactor was extrapolated from HFBR (3.6 %) taking into account the larger number of beam holes in our design. From this value, the effect for the H_2O cooled reactor was obtained using a homogenization method. The temperature defect was based on preliminary calculations but it is comparable to the value used for HFIR. The 1000 pcm margin allows the reactor to be restarted after an accidental shutdown. It is seen that the total excess reactivity required is about 1300 pcm below the provided excess reactivity, table 4. This reserve will be required for manufacturing tolerances in the fuel element, uncertainties in nuclear data, etc.

Table 6 REACTIVITY BALANCE

	D_2O reactor	H_2O reactor
Beam tubes	4500 pcm	2835 pcm
Remaining burnable poison	970 pcm	1863 pcm
Temperature defect	500 pcm	500 pcm
Margin at end of core life	1000 pcm	1000 pcm
Xe, Sm	5000 pcm	5000 pcm
U burnup and Stable F.P.	9820 pcm	9170 pcm
Total	21790 pcm	20368 pcm

5. EXPERIMENTAL FACILITIES

5.1. BEAM TUBES

Table 7 lists the main beam tubes as they are presently envisaged, (cf. also Fig. 2). The hot and cold neutron source are introduced from above through cylindrical tubes. The bunch of neutron conductor tubes to be used in connection with the cold neutron source is described in another

paper at this conference [10]. There will be some additional small vertical or slightly inclined thimbles, entering the reflector tank from the pool surface, for irradiation purposes.

Tabel 7 BEAM TUBES SPECIFICATIONS

Number	Description	Diameter	Remarks
2	tangential channels	100 mm	not shown in Fig.2
1	inclined tangential channel	100 mm	
1	tangential channel	60 mm	
1	tangential channel	150 mm	
1	radial channel	100 mm	
3	tangential channels	100 mm	viewing hot neutron source
1	bunch of 5 neutron conductors	-	viewing cold neutron source
1	horizontal through tube	100 mm	not shown in Fig.2
1	inclined through tube	170 mm	

The exact axial, radial and angular positions of the beam tubes in the reflector are still object of a very detailed study which is presently under way, taking advantage of experience gained during the first HFBR measurements. The high flux region within the reflector for both solutions can be visualized in Fig. 8, showing that the volume where the flux is greater than $0.8 \cdot \phi_{\text{max}}$ is larger for the D_2O -cooled reactor. Calculations of the fast and epithermal neutron as well as the γ ray background were also performed. They indicate that at positions with comparable thermal neutron fluxes the backgrounds are lower in the D_2O -cooled version.

Each beam tube will be made replacable. This is required from the experimentators in order to attain flexibility, but it is also necessary due to high thermal stresses and to radiation damage in the beam tube tips (silicon production from thermal neutron capture in aluminium, helium and hydrogen production by n, α - and n, p -reactions of fast neutrons). To this end, various solutions have been found using flanged junctions with metallic gaskets or welding of the sections. A lead cask, providing about 12 cm of shielding, will be used during transfer of tubes. Replacement of the "multipod" for the cold and hot neutron source appears difficult if no wall between the source and the beam tubes is allowed, and studies on these experimental facilities are under way. Handling of the vertical beam tubes

will be from above, requiring sufficient space at the pool surface and above.

The shielding around each horizontal beam tube exit is removable, forming an "experimental niche". This facilitates access to the flanges which have to be opened if a beam tube is exchanged and adds considerably to the flexibility of the beam tube. The remaining wall between the niche and the pool is also removable and made from aluminium in order to reduce activation of the wall by neutrons leaking through the beam tubes. Collimators or other experimental equipment can be installed in the removable part of the beam tube shielding which also contains a beam shutter. Each beam tube will also be equipped with a fast-acting valve in order to suppress leakage of heavy water into the experimental hall in case of a beam tube rupture. Each beam tube can be evacuated, a high vacuum being required in some cases.

5.2. HOT NEUTRON SOURCE

A graphite block heated to about 2000° C will be used. Heating will be by core γ -rays and fast neutrons. An experimental and theoretical study is under way in order to find the optimum size, shape and position of the hot source. Using electrical resistance heating, a graphite cylinder 20 cm in diameter and height was heated to 1200° C. It was embedded into a reflector of cold D_2O about 1 m^3 large, and neutron spectra from the graphite surface were measured using the Karlsruhe pulsed source-time of flight facility with a 12 meter neutron flight path. Calculations of the spectra were attempted using the THERMOS code. Using realistic scattering kernels for both D_2O and graphite, fairly good agreement between measured and calculated spectra was found. The theoretical tools thus tested can now be applied to the actual hot source design problem. We expect to reach a "gain factor" of about 30 at an energy of 0.3 eV. Parallel to these neutronic studies, technological studies as to properties of hot graphite, heat removal, handling of the source etc. are being carried out. As an intermediate step, a graphite hot neutron source will be installed in the Karlsruhe FR 2 research reactor in 1967. Other materials for the hot source, i.e. beryllium oxide and cerium hydride, are also being considered but only very preliminary studies have been made so far.

5.3. COLD NEUTRON SOURCE

Two concepts for the cold neutron source are being considered: A vortex-type source using liquid hydrogen, and a "conventional" source using sub-

cooled liquid hydrogen (or deuterium) which is circulated and cooled in an external heat exchanger. In the vortex concept [1], a rotating layer of liquid hydrogen is formed on the inside of a cylinder by injection from a proper system of nozzles. Out-of-pile tests of this concept have been carried out, demonstrating that a stable layer of proper thickness can be obtained. A programme was started to investigate the neutronic properties of cold neutron sources, especially to define their size, position and composition. To this end, a D_2O reflector about 1 m^3 in size was installed at the SILOETTE swimming pool reactor of CEN Grenoble. Liquid hydrogen or deuterium vessels of varying geometries can be placed within this reflector, and neutron spectra are measured by the time-of-flight method using a chopper. Spectrum calculations are carried out using the THERMOS code. Preliminary results lead us to expect that the highest gain in cold neutrons can be obtained with a large (40 cm diameter) liquid deuterium source. However, the heating of this source would be considerable (about 18 KW). This heat had to be removed by the external heat exchanger. With the vortex source, the heat rate would be lower and the heat removal simpler since hydrogen boiling occurs and no heat exchanger is needed.

Recently, H. MAIER-LEIBNITZ has suggested to study "cool" neutron sources, consisting of hydrogen gas at a high pressure and at temperatures ranging between 20 and 100°K .

6. CONCLUDING REMARKS

Our study has revealed that the design objectives can be met by both the H_2O - and the D_2O -cooled version. There are many features common to both versions. There are, however, characteristic differences: The H_2O -version has a simpler cooling circuit, offers simpler fuel handling and is less expensive, it will also be somewhat simpler to operate. The D_2O -version offers a larger high-flux volume available for experiments and somewhat lower backgrounds. The fuel is at a lower temperature level, making a later increase of flux and power possible, and the problem of the core shroud is less severe. The fuel cycle costs will be slightly lower, partly due to the lower power, partly due to the longer duration of the fuel cycle. On the basis of the results of this study, the steering committee of the project will make a decision on the coolant medium within the very near future. After this the preliminary design can be rapidly completed and it is hoped to pass some design contracts to industry in the not too distant future. Scheduled completion date for the reactor is late 1971.

A fairly broad experimental programme will have to be carried out during the design and construction phases. This includes heat transfer studies which are already carried out at Grenoble and will be extended in the future. It furthermore includes the neutronic and technological studies for the cold and hot neutron source which have been started at Grenoble and Karlsruhe. Critical experiments have been started at ALIZE III at Saclay which is a light-water moderated, heavy-water reflected assembly and the construction of a further critical assembly is being considered. At Grenoble, an experimental programme to study aqueous corrosion on newly developed cladding materials is under way. Further experimental programmes presently under consideration concern structural and fuel material irradiation, fuel element manufacture and hydraulic testing.

Detailed design, construction as well as all these experimental programmes will be carried out in collaboration between several laboratories and industrial groups within the two countries. Building the Franco-German High Flux Reactor will thus be an interesting experiment in international collaboration on a specific large project. We believe that the experience gathered during the past phase of preliminary studies has been most encouraging.

7. ACKNOWLEDGEMENTS

It is impossible to name here all those individuals who have taken part in the work and the authors take the opportunity to express their thanks to all those who have made contributions to this report. However, we would like to express our special gratitude to Dr. H.A. Kouts (BNL) and the team of the Brookhaven High Flux Beam Reactor for the very generous supply of information obtained in their project. Also, we would like to thank Professor H. Maier-Leibnitz, the designated director of the Franco-German institute, for his invaluable support of the project.

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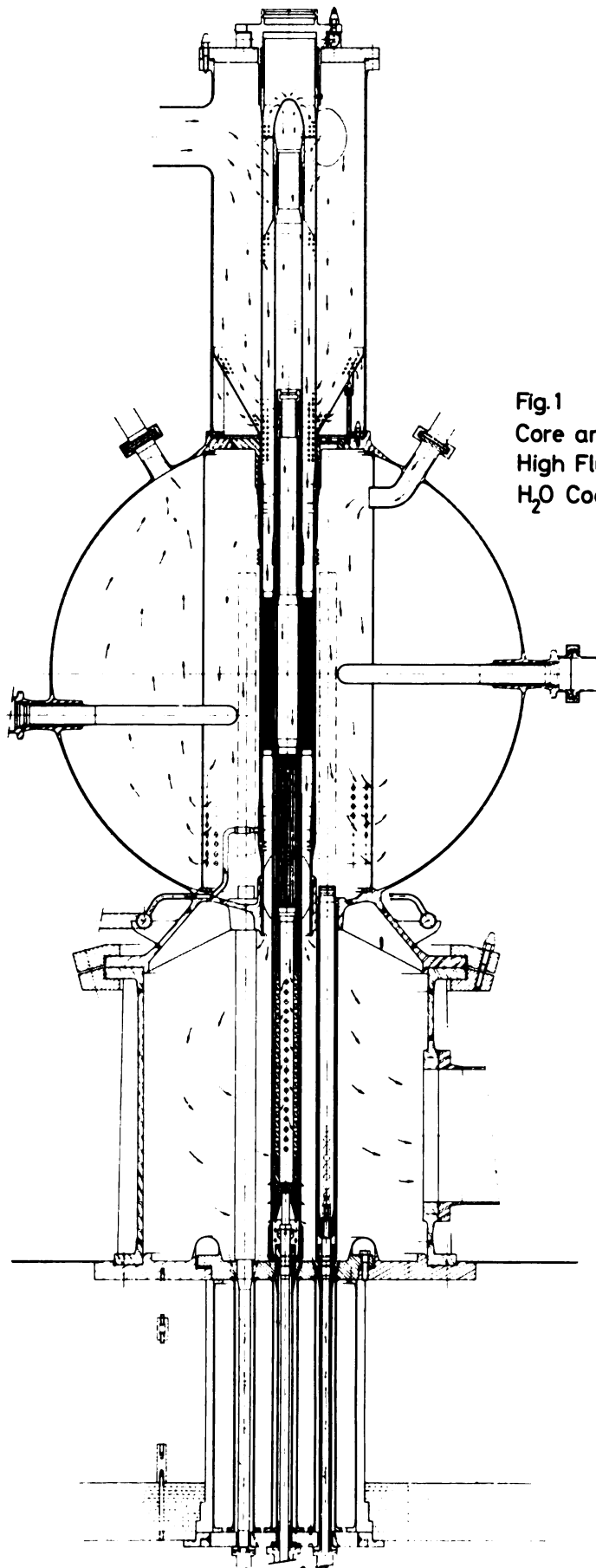


Fig.1
Core and Reflector of
High Flux Reactor,
H₂O Cooled Version

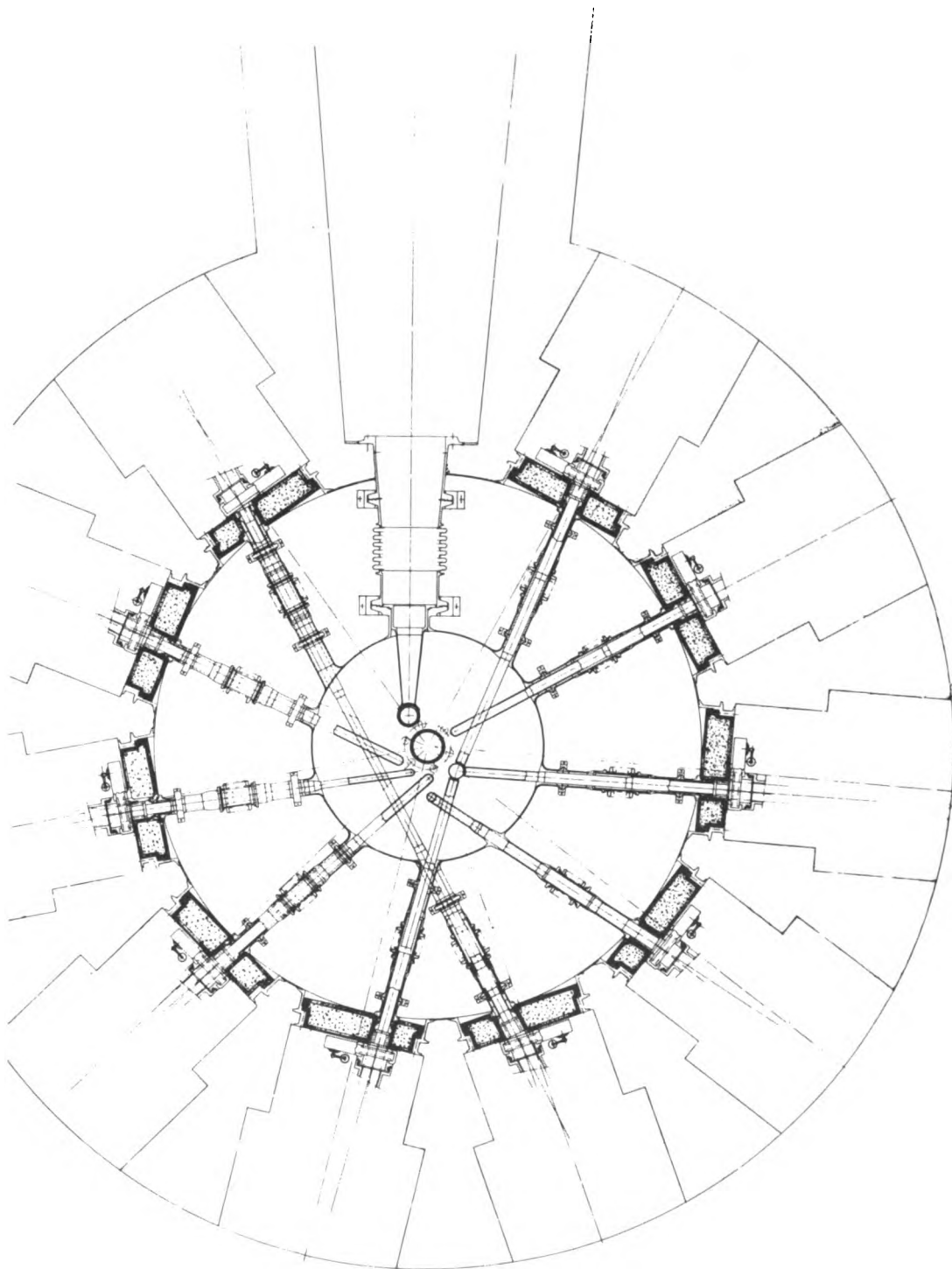


Fig. 2
Schematic Horizontal Cross Section of the High Flux Reactor

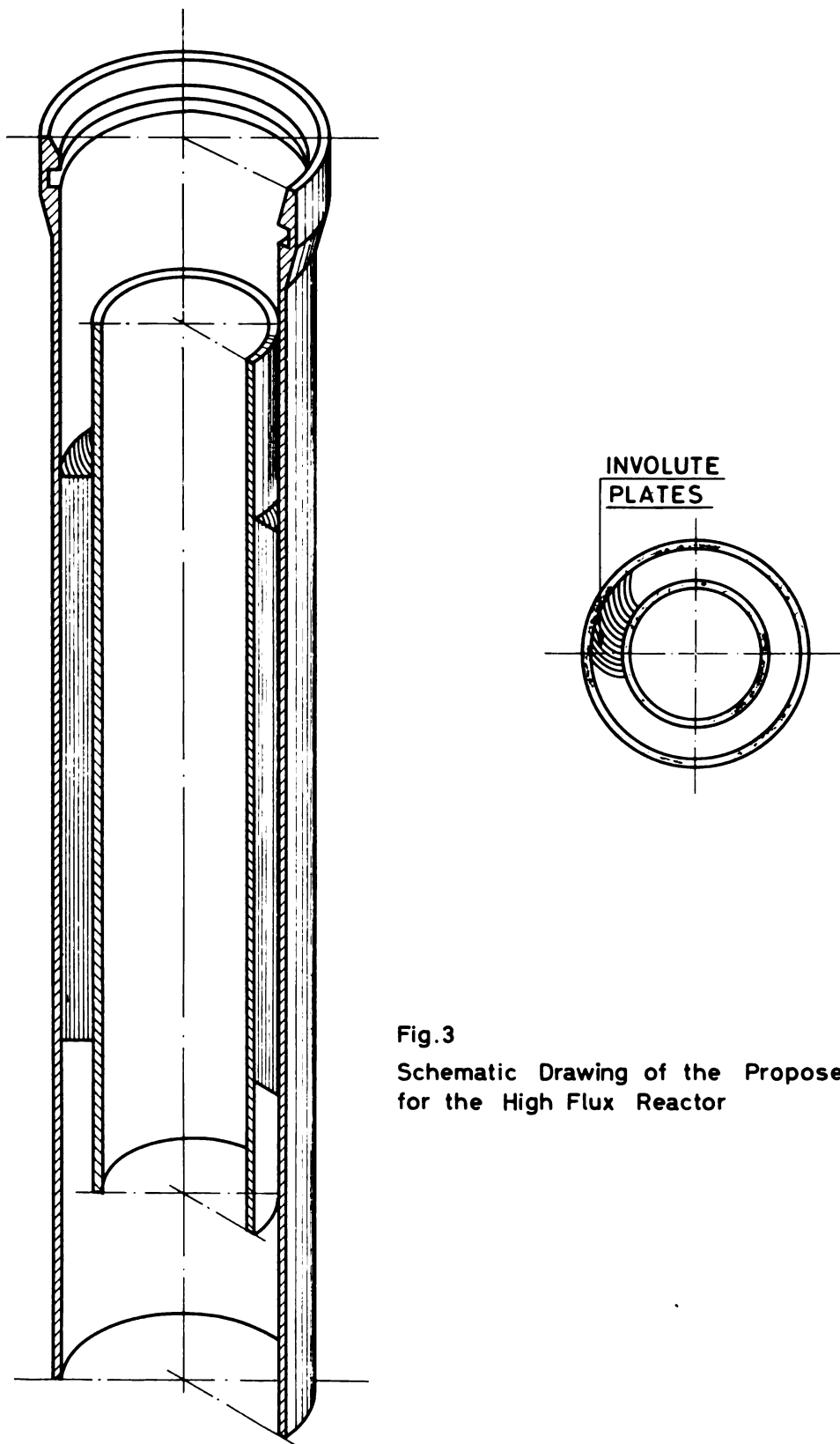


Fig.3

Schematic Drawing of the Proposed Fuel Element
for the High Flux Reactor

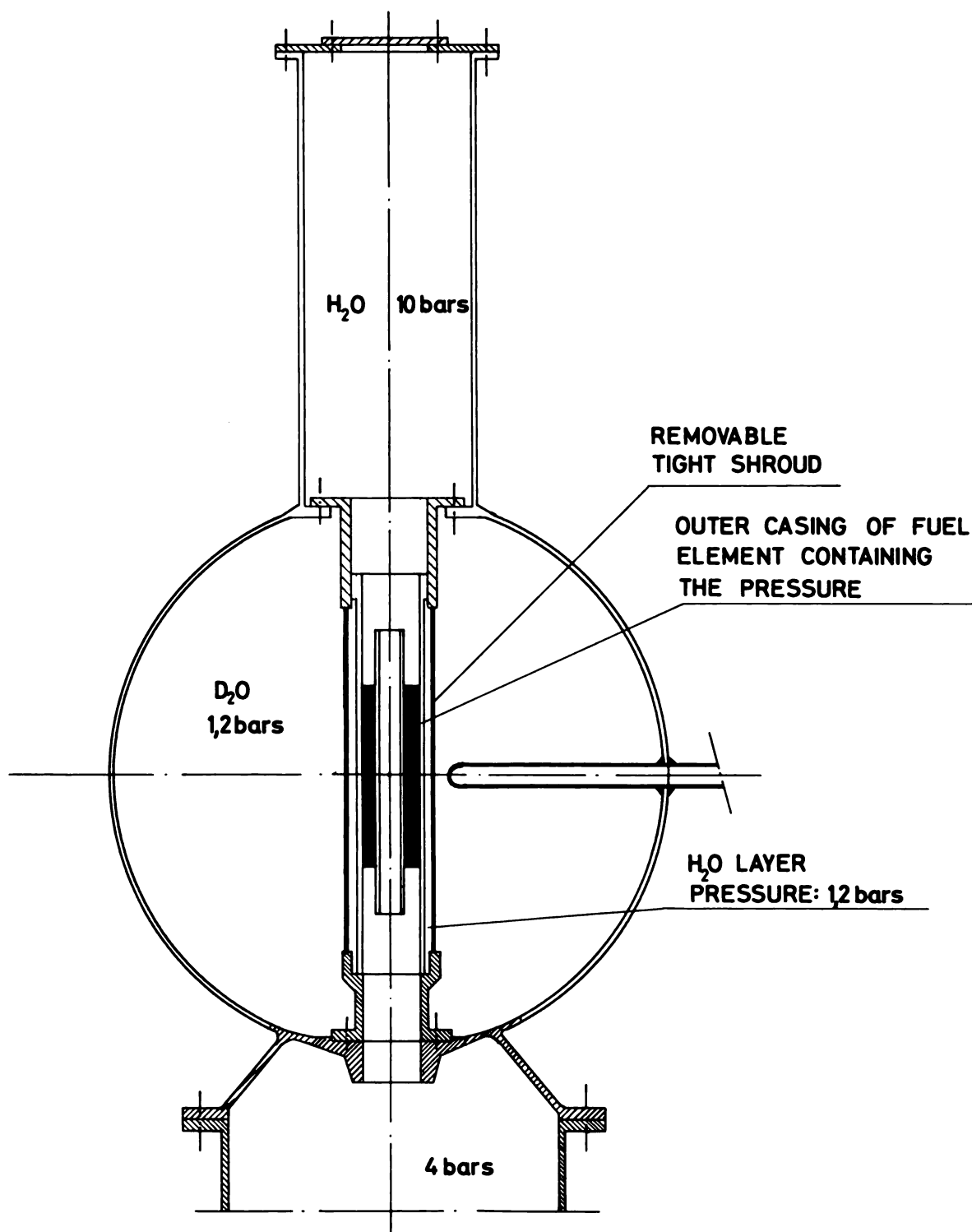


Fig. 4: Schematic Drawing of the Core-Reflector Separation Arrangement for a H₂O-Cooled High Flux Reactor

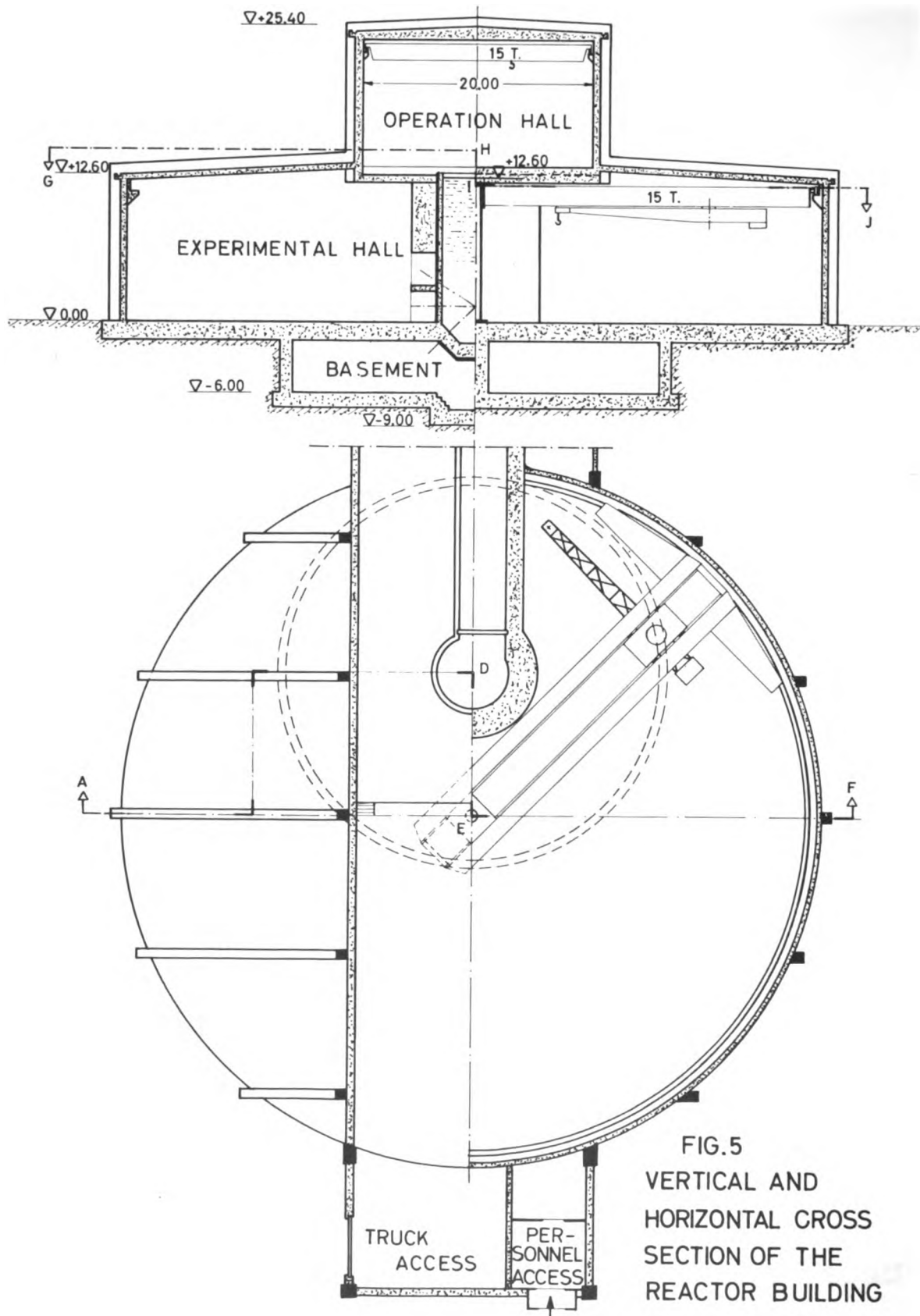


FIG.5
VERTICAL AND
HORIZONTAL CROSS
SECTION OF THE
REACTOR BUILDING

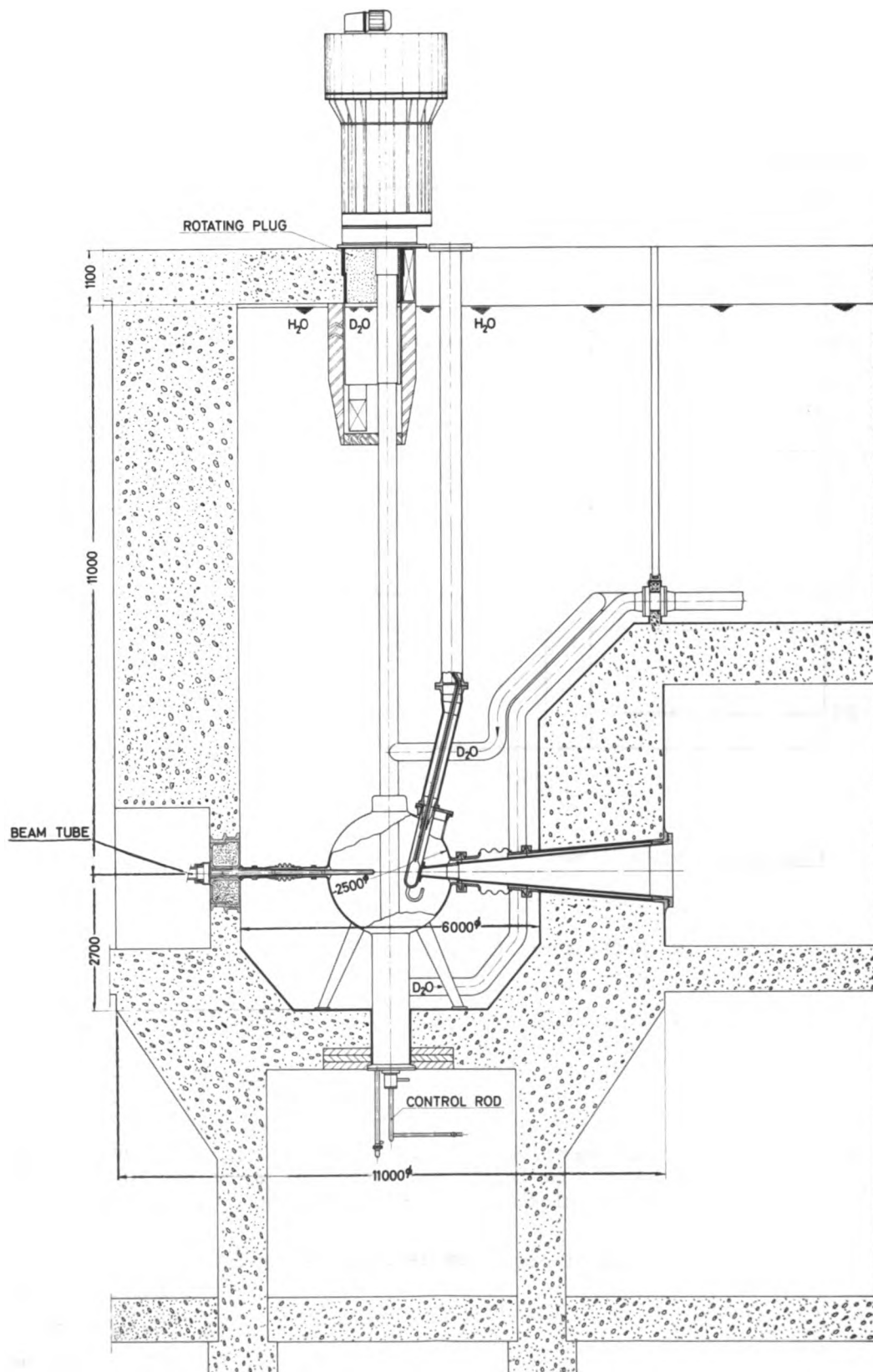


Fig. 6: Vertical Cross Section of the High Flux Reactor, D₂O Cooled Version with Rotating Plug

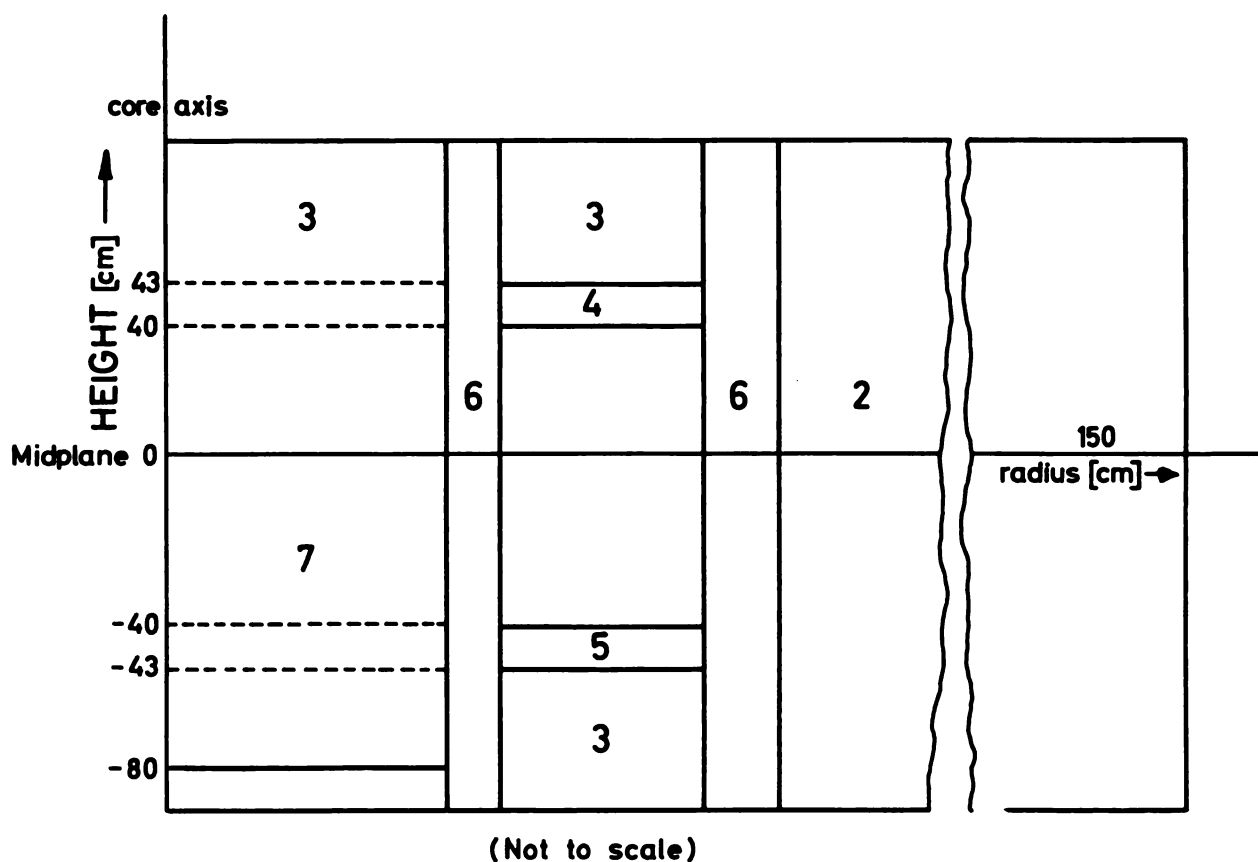


Fig. 7 a: Composition of H₂O Core

1 Core: 185 g U₅/1

$$\frac{\text{Vol. metal}}{\text{Vol. H}_2\text{O}} = 0.706$$

2 Reflector: D₂O with 0.2% H₂O

3 Reflector: H₂O

4 Ends of plates: Al

$$\frac{\text{Vol. Al}}{\text{Vol. H}_2\text{O}} = 0.706$$

5 Ends of plates: Ni

$$\frac{\text{Vol. Ni}}{\text{Vol. H}_2\text{O}} = 0.706$$

6 Structures: 1/3 H₂O, 2/3 Al, representing at the exterior a 6 mm thick Al fuel element casing, two zircaloy shrouds each 4 mm thick, and water layers; at the interior, the 6 mm thick Al inner casing of the fuel element, and water layers

7 Control rod: 80% Beryllium + 20% H₂O

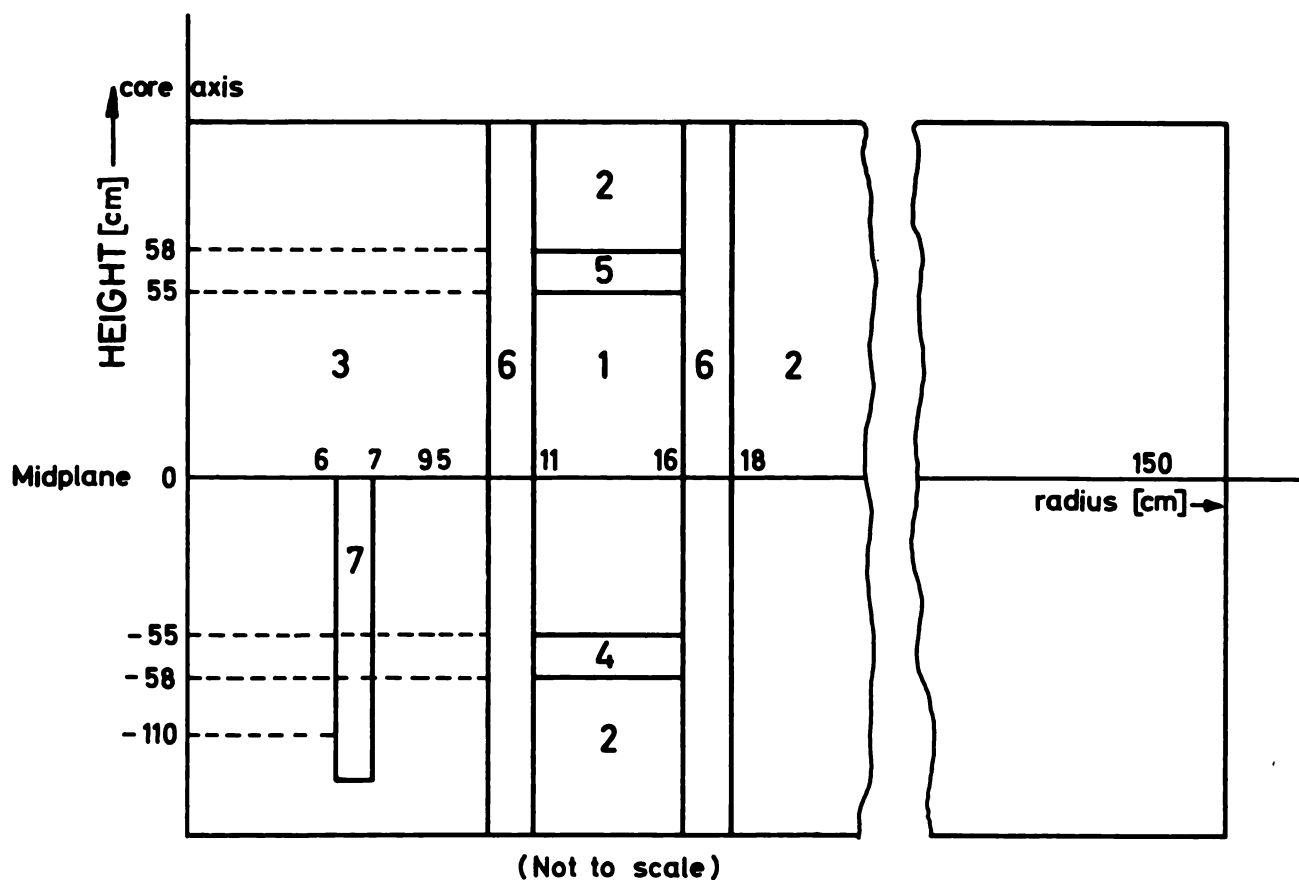


Fig. 7 b: Composition of D₂O Core

- 1 Core: 185 gU₅/l $\frac{\text{Vol. Metal}}{\text{Vol. D}_2\text{O}} = 0.706$
- 2 Reflectors; D₂O with 0.2% H₂O
- 3 Central hole; 95% D₂O + 5% Al, representing structures such as rod guides
- 4 Ends of plates; 60% D₂O + 40% Al
- 5 Ends of plates with boron; 60% D₂O + 40% Al + boron
- 6 Structures; 60% D₂O + 40% Al, representing at the exterior a 6 mm thick Al fuel casing, a shroud of 4 mm of zircaloy, and water layers; at the interior, 6 mm of Al
- 7 Nickel control rod; 70% Ni + 30% D₂O

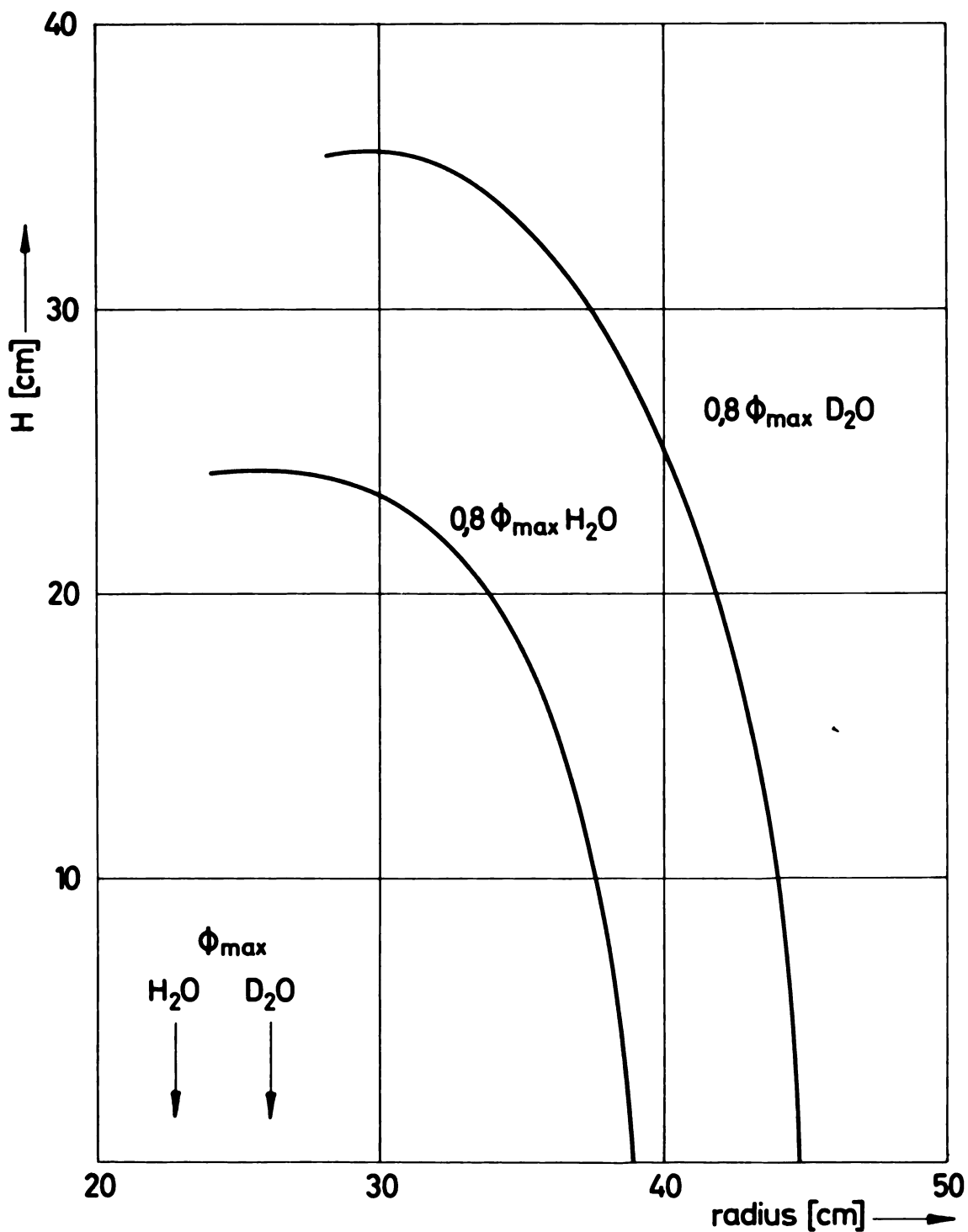


Fig. 8: Regions in Reflector where the Thermal Neutron Flux exceeds $8 \cdot 10^{14} \text{ cm}^{-2} \text{ sec}^{-1}$. The Flux Maximum is in the Midplane at the Points indicated by Arrows

DISCUSSION
OF
PAPERS II.A.1-6 CONTINUOUS REACTORS

Chairman: K. H. Beckurts

Secretary: C. Anderson

COLE: I would add a couple of comments in regard to the HFIR which simply did not get in Dautray's paper because of the time scheduling. We started operations at 100 MW on September 9 and as of the present time have run about half of a core cycle at the 100 MW power level. We also, of course, did run the preceding cycles of 75 and 90 MW. The reactor seems to have behaved quite well. I think I have no further comment on the HFIR, but I do have a couple of questions which I will hold until other authors have had an opportunity to comment.

SHAFTMAN: I would like to make a remark on the concept of the steady-state power limit and the ultimate that one would expect out of such a continuous reactor. Naturally, if one designs to be safe at steady-state power one might simply be taking a chance if there were some reactivity incident, so one has to be especially careful, keeping in mind what might happen in the way of a transient; you would like to design so as to avoid a melt, but you do not want to be too conservative in your design. The other comment I wanted to make about the AARR in particular is that, really, a reactor of this type is sized so that it would be better at a higher power level, and not at 100 MW. The reason for this is that we have a conservative safety margin for burnout, even keeping in mind this transient phenomenon I mentioned. We are reactivity limited there in that, with a system as heavily loaded as this, we have to make the core somewhat thicker than we otherwise would like to, so as to get the core life that we are trying to get (90 days at 100 MW or roughly 45 days if we were to go to as much as 200 MW). I would just like to take note of Fig. 15 in the paper on AARR which shows the relative contribution of source neutrons to the peak flux in the beryllium reflector. There is a similar figure for the internal thermal column which shows that it would be very nice (and this is well known) if you could put your source neutrons very close to your receptor, but that is difficult to do in a reactor and still operate it in a practical way. Figure 16 shows the effects of making the reactor thinner. The two effects are that we could make the core somewhat thinner radially (the annulus thinner) and we could get a substantial increase (but much less than a factor of two) in the peak flux per unit power. But this costs quite a bit in reactivity and I think this is generally the case, not only for a reactor of this type,

but for other reactors, small reactors, where leakage is depended upon to produce the high fluxes for the users.

COLE: It appears that in the measurement of the neutron current in the Brookhaven high flux beam reactor there seems to be at least a factor of two available in the way that they measure this. I would like to ask a question of Kouts: Does this not perhaps give one a potential lead to improving beam qualities, if you wish to call it that, by judiciously choosing reflecting surfaces, etc., at least for certain types of experiments where this would be consistent with the needed degree of collimation?

KOUTS: Cole is referring to the difference between some of our earlier measurements of beam intensity and some of the later measurements, the difference being the fact that we use fine collimation in order to view the inner ends of beam tubes and in the earlier experiments had considerable amount of neutron reflection from the collimators, that is, from the walls of the collimators. This led to an exit current in early measurements larger by a factor of two than those in later experiments. This, indeed, does lead to a possibility of increasing the intensity available in certain experiments, although of course there is a limitation in that you destroy collimation and only with certain experiments is it possible to avoid this as a problem. I thoroughly agree with you.

SHAFITMAN: I would like to direct a question to Cole about this concept of Cheverton's of going to an island flux of 3×10^{16} unperturbed flux. Is this an H_2O island with D_2O ?

COLE: Yes.

BECKURTS: May I ask Cole, have you an idea what would be the annual fuel cycle cost in this 3×10^{16} operation?

COLE: Actually, what I had stated in my paper is slightly different from the statement by Dautray, just due to the time sequence of the papers coming out. I had made the comment in my own paper that, while this seemed to represent a reasonable upper limit for reactors of this class, at the present time it did not appear to be practical either from the pumping power considerations or from fuel cycle costs. I can not give a figure on fuel cycle cost. It would be extremely high.

BECKURTS: It would be about 6 to 8×10^{15} in the reflector, is that right?

COLE: Of this order, yes. This would be a 750-MW, heavy water-cooled and reflected reactor, utilizing a light water island.

BECKURTS: This seems to be consistent with Dautray's statement. In reactors of the present generation we have about 2×10^{15} reflector flux. A factor of two is possible with slight modifications and another factor of two is possible with further modifica-

tions. That would lead to about 8×10^{15} . But I think one should be aware that the annual fuel cycle cost would increase by more than a factor of 4.

COLE: Yes, much more.

KEILBER: I just want to point out that at these very high power densities the core cycle obviously decreases and shutdown times become important in considering the mean flux for any long term experiment.

PERRY: In answer to your question, Dr. Beckurts, I have some figures here which were calculated a year ago at the occasion of the SOC-ING-UNS Conference in Gatlinburg. I calculated at that time that this 750 MW reactor that Cheverton sketched out would be capable of being loaded for about a four-week fuel cycle. On a four-week fueling cycle the fuel cost would be about \$6 million a year, of which about \$4 to \$4-1/2 million would be burnup costs for the ^{235}U . I think that most people feel that for a power reactor of such a high power level it would be very desirable to find ways of recovering some of the heat costs. It is perhaps worth noting in passing that this \$6 million a year corresponds to 20¢ per million BTU's and one might expect to recover some portion of that cost.

KUCHLE: Stimulated by the calculations of the Franco-German reactor, Dalle-Donne and Kallfelz in our laboratory made a short estimate on a reactor that still contains the basic features of the Franco-German reactor, but where the power density is pushed to the ultimate limit. The intention was to see what would be the engineering consequences and operating costs of the reactor that is about the same flux in a similar geometry as would be produced by the ING. It turned out the limiting factor was the thermal stress in the clad and not so much thermodynamics. Therefore, the choice of the coolant was made to obtain a minimum temperature and this coolant was not sodium but super critical water. We had a stainless steel core with Inconel-625 clad. Referring to Table I, the central point is that it is a reactor of about 500 MW, which can produce a flux of about 10^{16} with a fuel cycle time of around six days. We would consider this as not a very attractive approach, in agreement with what was said here, since fuel cycle costs would be very high. Instead one should better look for an approach where the fuel fabrication costs are drastically reduced and thus, in our opinion, some way of a homogeneous fuel cooled outside the reactor seems to be the most promising approach. The geometry is similar somewhat to the Franco-German; it has a central beryllium flux trap, a one-ring HFIR core with radial power flattening, stainless steel core, and Inconel cladding. The study on corrosion that was made by G.E. shows that the corrosion would not be a serious problem even under the conditions of this reactor. The main point is that the thermal problems in the clad provide the limitations.

BECKURTS: We could finish perhaps with this part of the discussion here. There seems

to be a certain coherence of opinion on how far one could go with water technology. In many cases it is possible to reach higher fluxes but it would be very expensive, and I think one has to think very much on the maximum operation costs one can justify in view of experimental programs. This is not an easy task. Perhaps we could consider some other topics. For instance, would somebody like to comment on the question of background in these reactors? What effort should be made concerning backgrounds? There seems to be two different schools. Apparently there is the beryllium school and the heavy water reflector school, and apparently the background situation will be different in each.

SCHULT: It is difficult to consider the background of the reactor in general. What you usually do is think of a particular experiment and you realize that you have some kind of a background, then you wonder what can be done to decrease this kind of background. One can almost make crazy approaches; one would be, I guess, to think of a gamma shield around the reactor core. I do not know what will happen to the neutron flux and so on. Reactor designers are much more experienced to find a reasonable answer. In cases where one works with cold sources, for instance, the cooling problems (especially if the sources are large) would be less difficult. On the other hand, my feeling from the last discussion about high flux reactors is almost this: the reactor is something like a black box, and one talks about high neutron fluxes. One does not talk about other possibilities to improve what we, in principle, want to have, namely, data. And these improvements can be done either by brute force, which means high neutron flux, or by a little more sophisticated technique (sophisticated in the sense of, maybe, the Apollo program), and this sophistication means you really sit down and think how much money you have to put into your experiments to get better data. So I think it will be a really difficult problem to find a solution between a high, ultra, or super high flux reactor and a set of experiments which in total give you a good result. Also, to compare high flux reactors one would also wish to have, maybe, something of a figure of merit and then the question is how could one, for instance, get one of these figures? One could think of doing the following things: You use the number of beam tubes you have; you consider the flux you have in these beam tubes; you use some kind of a weighing factor to take into account the experiments you want to perform with these beam tubes; you make a summation of the whole thing and you divide this by the cost of your reactor. Then you know whether you can do scientific work expensively or inexpensively.

KLEY: I have two questions. The first is: What gain factor do you really expect in the Franco-German high flux reactor from the cold source? Although it is relatively large, the thickness is very much reduced, being only about 1 cm. So, what gain factor can you anticipate from this source? The second question is: You use hydrogen and you have a composition of ortho- and para-hydrogen, and as you know

very well the cross section is changing from the ortho- to the para-hydrogen concentration; do you foresee a continuous control of the mixture? How do you solve this problem?

BECKURTS: First, I would like to say it is by no means sure that the vortex source, which has this 1 cm layer, will be used, but the experimental program includes also a study of large sources, and I think Dautray mentioned that a big 40 cm deuterium source is under experimental study. I do not remember the gain factors.

AGERON: It is impossible to give a straight figure for an experiment that has not yet been done, but a large deuterium source compared to an optimized hydrogen source will give roughly a factor of two at 5 Å and a factor of three at 10 Å. For the absolute value compared to an ordinary heavy water moderator at ordinary temperature, the gain factor will be roughly 10 for hydrogen and 20 for a large deuterium source at 5 Å, and roughly 20 for hydrogen at 10 Å and perhaps 100 for deuterium.

SPINRAD: I would like to inquire further as to the technical limits on the engineering and provision of these cryogenic sources, and my motive is one of skepticism that they are so expensive. Many years ago, when we were still proposing a Mighty Mouse reactor at Argonne, the question came up as to whether the internal thermal column could not be made out of a cryogenic zone consisting of perhaps heavy ice, frozen deuterium or (the last thing that we tried to get rid of coherent scattering) pressurized helium. We called in consultants from some of the cryogenic firms to inquire about the costs, and provided them with expected heat loads. We were surprised how modest the figures would be to extract a few hundred kW of heat at 4°K; they did not seem bad at all. The cost now seems to have come up to be a major factor, and I would like to inquire how come?

BECKURTS: I would just like to add that those people who are now thinking about super-conducting linacs are considering removal of many hundreds of kW at liquid helium temperatures. They seem to be optimistic about it.

JACROT: You have also to consider the other aspect which is that you have an intrinsic limitation which is the amount of heat which you can extract at the level of your hydrogen or any kind of cold device. This is the same limitation that you have on the uranium in the reactor, which is the number of watts/cm² that you can extract. This is a physical limit beyond which you cannot go extremely far. So it has to be considered if, in a reactor with a flux of a few $\times 10^{15}$, it would be feasible to make a cold source. But I agree with you it is a fact that the price of the device is not increasing as the power but is increasing much more slowly; that is quite sure.

KLEY: The safety problems you have in the deuterium case really amaze me. I think it is very nice to think of using deuterium as a cold source; but I think you have an additional problem, a safety problem. The amount of deuterium you have in your reactor

is probably of the order of 10 liters. Now, what are the considerations as far as safety factors are concerned for this cold source of either hydrogen or deuterium? Because in the case of hydrogen, you have much less amount of liquid hydrogen in the reactor.

AGERON: I understand that the safety problems arise from the quantity of liquid cryogenic in the reactor. It seems that, effectively, there is a very large difference between a deuterium source (we will need roughly 40 liters of liquid deuterium) and a hydrogen source (which will need only about 1 liter), but is an unanswered question from what comes the danger. Is it in the liquid form or is it in the gaseous form? For instance, we hope that a liquid deuterium source will be entirely sub-cooled and, if the danger is from gas, the safety problems will not be much larger than for the hydrogen source.

MAIER-LEIBNITZ: I would like to raise one question about which some of you may have some experience, namely, how constant do you think you can keep your neutron source from a reactor? When you have a large background, what you can measure depends to a very large degree on whether you can detect, let us say, 1% above the background. It means, in this case, the flux should not show irregular fluctuations of that order.

I would very much like to take this opportunity to say that the Franco-German reactor project owes very much to one of our hosts here, the ENEA, and especially to Prof. Kowarski who is present here and who first had the idea of a high flux reactor in Europe.

COLE: I think that, before making a comment on how constant one can hold flux, I have to ask a question over what period of time you are willing to integrate, because you can hold it very constant over a period of, let us say, one hour or even one minute. But if you start looking at it in, let us say, several cycles per second or so, then the fluctuations get to be in the order of a fraction of a percent, but can be held that closely in a well-designed high-power density reactor. The higher the power density the more difficult this seems to be because of temperature excitation of the various instabilities in the reactor. But, I would say, one can certainly hold to the order of 1% on a time schedule of a minute or longer. To hold it much longer than an hour also becomes difficult, and you probably cannot hold it to 1% averaged over a day. One could do it for one experiment over a certain period of time if the operators were instructed that this was important. The nominal variation is of the order of, maybe, 2%. The operators would have to have a monitor at the location of your experiment, of course.

SHAFTMAN: I would like to comment on this also. Our users have told us that for AARR they do not want to favor one particular beam tube. I raise the question as to whether it would be desirable to hold the flux at one beam tube constant, which is not necessarily the same as holding the reactor power constant. But, of course, if

you have a concern about what the flux level is at the beam tube I do not see any reason why you could not have a local monitor which would tell you that. So, perhaps, this would solve the question, without the necessity of holding the power constant.

TABLE I
Preliminary design characteristics of a 10^{16} -flux beam research reactor

<u>1. Dimensions and composition</u>		<u>2. Operating characteristics</u>	
<u>D₂O-reflector:</u>	diameter 300 cm height 200 cm pressure near atmospheric	Coolant velocity	32 m/sec
		Coolant temperature at core entry	40°C
<u>H₂O-core:</u>	diameter 27 cm height 100 cm central flux trap filled with Be, diameter 14 cm core volume 42 l	Coolant temperature at core exit	160°C
		Pressure at core entry	300 Atm.
		Pressure at core exit	250 Atm.
		Maximum heat flux	2500 W/cm ²
		Average heat flux	1180 W/cm ²
<u>Zr-pressure barrel</u>	between core and re- flector, thickness 2.0 cm	Maximum wall temperature of fuel plate	590°C
<u>Fuel element:</u>	one ring HFIR-type with radial power flattening	Maximum power density	25 MW/l
		Average power density	11.9 MW/l
		Total power	500 MW
<u>meat:</u>	stainless steel 16/13 with 40% fully enriched UO ₂	Maximum unperturbed flux in the reflector	$1.0 \cdot 10^{16}$ /cm ² sec
<u>clad:</u>	INCONEL 625	k _{eff} of fresh core	1.22
	coolant channel width 1.2 mm	Length of cycle	6 d
	average fuel load 680 235U/l	Average burn-up	12%
<u>Coolant:</u>	supercritical water	Fuel cycle cost per year	20 · 10 ⁶ \$

To be presented at the
Symposium on Intense Neutron Sources,
Santa Fe, New Mexico,
September 19-23, 1966.

LIMITATIONS OF STEADY STATE, HIGH FLUX REACTORS,
CURRENT AND FUTURE*

by

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* Work performed under the auspices of the U. S. Atomic Energy Commission.

I. INTRODUCTION

The current generation of high flux research reactors⁽¹⁾ represents the penultimate in their performance, according to their designers. That is, while they may be improved by learning more about the actual system performance with regard to hot channel factors, coolant flow patterns, burnout safety margins, and so on, it is considered unlikely that this improvement could lead to more than a factor of about three in available flux. (If this were not the case, it goes without saying that these improvements would be the subject of vigorous research and development, and would be incorporated in the most recent proposals.)

These reactors, heterogeneous and with water coolant and moderator (light and heavy), have projected beam and irradiation fluxes of $1-5 \times 10^{15}$ thermal neutrons/cm²-sec.

The major factor limiting further advance is restricted power density, which is controlled by conflicting requirements of heat transfer, heat transport, and core materials. Current designs have coolant flow velocities of 6 to 15 mps, core pressure drops of 5 to 10 atmospheres, and core power densities of 1 to 3 MW/l. Typical heat transfer rates are of the order of 200 W/cm² and system pressures are tens of atmospheres to suppress surface boiling.

To go further, therefore, it is necessary either to overpower the limitations of current reactor types or to find new types of systems more amenable to very high flux design. In either case, we must reexamine the basic principles of research reactors in order to progress.

II. GENERAL CONSIDERATIONS

High neutron flux results from a combination of three factors:

1. High neutron production rate in the source
2. High neutron availability from the source
3. Low neutron absorption in and leakage from the experimental region.

A Production Rate

In a steady state reactor, the first of these factors is directly propor-

tional to the reactor power and to the value of ν (neutrons/fission) in the reactor fuel. In fact, if

$$\begin{cases} Q & \equiv \text{neutrons produced/second} \\ P & = \text{power in MW} \end{cases} \quad (1)$$

$$Q \equiv 3.25 \times 10^{16} \nu P \quad (2)$$

Thus, other things being equal, the production of high flux per unit power is enhanced by using fuel of high ν . The three standard fissile isotopes are characterized by values of ν and η (neutrons/fuel absorption) as listed in Table I, and it is evident that Pu^{239} is superior on this count. (The values in Table I are very crude, but sufficient for inspection.)

Table I						
Fission Parameters of the Fissile Isotopes						
Energy Range	U^{233}		U^{235}		Pu^{239}	
	ν	η	ν	η	ν	η
Thermal	2.50	2.29	2.43	2.07	2.87	2.00
Intermediate	2.51	2.25	2.44	1.65	2.89	1.80
Fast	2.60	2.37	2.51	2.28	3.00	2.73

B. Neutron Availability

When neutrons are extracted, they must be neutrons which are not needed to maintain the chain reaction. Let us consider, then, how many neutrons are available.

Per fission yielding " ν " neutrons, the critical reactor core absorbs ν/k where k is the infinite multiplication factor of the core. Therefore, the excess neutrons produced per fission are $\nu(k-1)/k$.

Barring parasitic absorption, the maximum value which k can assume is η . Table II therefore lists values of $\nu(\eta-1)/\eta$, which represents the maximum num-

ber of neutrons which can be made available, per fission.

Table II			
Available Neutrons per Fission ($\nu[\eta-1]/\eta$)			
Energy Range	U ²³³	U ²³⁵	Pu ²³⁹
Thermal	1.41	1.26	1.43
Intermediate	1.39	0.96	1.28
Fast	1.50	1.41	1.90

From Table II, it is apparent that the great superiority of Pu²³⁹ remains only in the fast energy range. Uranium-233 is the best fuel in the intermediate energy range, and Pu²³⁹ and U²³³ are about equally good in thermal reactors.

Unfortunately, U²³⁵ is the most available fuel, and would be used preferably. Its use in an intermediate reactor core results in only half as many neutrons being available as in a Pu-fueled fast core, and 2/3 as many as in a U²³³- or Pu²³⁹-fueled thermal reactor. Uranium-233 is currently the least available fuel. It would appear worthwhile, therefore, to explore the possibility of using Pu²³⁹, which is plentiful, to fuel research reactors. The desirability of Pu²³⁹ is enhanced by its high neutron absorption cross section at all energies, which makes it rather easy for the infinite multiplication constant to approach η .

C. Geometrical Considerations in the Experimental Region (External Experiments)

The standard arrangement of a beam reactor consists of a spherical (or approximately spherical) reactor core immersed in a moderator, in which moderator are located experimental irradiation facilities or beam ports. The optimum location of these ports is approximately one root-neutron-age from the core, so as to diminish the background of high energy neutrons.

The neutron current at the experimental point is then

$$J = \frac{Q(k-1)/k}{4\pi(a + \sqrt{\tau})^2} \quad (3)$$

where a is the core radius.

A moderating medium characterized by a thermal diffusion area L^2 and diffusion coefficient D will have a flux to current ratio of

$$\frac{\phi}{J} = \frac{Lr}{D(L+r)} \quad (4)$$

at any radius r . Setting $r = a + \sqrt{\tau}$ and substituting (4) into (3) yields

$$\phi = \frac{Q(k-1)}{4 \pi k} \frac{L}{(a+\sqrt{\tau})(a+L+\sqrt{\tau})D} \quad (5)$$

Table III lists the flux-to-source factor, $\frac{L}{(a+\sqrt{\tau})(a+L+\sqrt{\tau})D}$ for various moderators.

Table III					
Moderator Flux-to-Source Factors					
Moderator:	H ₂ O	D ₂ O (Std. Quality)	C	Be	BeO
Factor:	Thermal Flux at "a" per available source neutron for the various moderators, assuming core radius of "a - $\sqrt{\tau}$."				
a(cm)					
5	0.15	0.0635	0.031	0.078	0.0835
10	0.073	0.0465	0.024	0.051	0.058
15	0.043	0.036	0.0195	0.0365	0.043
20	0.028	0.029	0.016	0.0275	0.033
25	0.020	0.025	0.014	0.021	0.0275
30	0.015	0.0205	0.012	0.017	0.0215
40	0.0092	0.0155	0.00905	0.0115	0.015
50	0.00625	0.012	0.00715	0.00855	0.0115
60	0.0045	0.00985	0.0058	0.0065	0.0088
The constants used for the various moderators are ⁽²⁾ :					
$\sqrt{\tau}$ (cm)	5.2	11.3	21.0	10.0	11.5
L (cm)	2.8	100.0	54.0	20.0	30.0
D (cm)	0.14	0.83	0.85	0.49	0.47

It is notable from Table III that H₂O is the best moderator for small cores, giving way, first, to BeO and then to D₂O as the cores get larger (the advantage of BeO is problematic, as the constants for it are poorly known).

It must also be emphasized that the numbers of Table III are somewhat optimistic. At one root age from the source, there is incomplete thermalization. Thus, the numbers are more indicative of total flux, varying proportions of which are thermal. Likewise, attenuation by absorption, which is of importance, particularly when H₂O is the experimental medium, has been ignored.

Small cores are limited in power output by power density, whereas with large cores the limit is total power. This matter will be discussed later. However, one facet of this argument is worth noting here. This is that, if engineering puts a limit on power density and cost puts a limit on power, the maximum flux is obtained when the two limits are achieved simultaneously. Then the proper core volume is

$$\frac{4\pi}{3} a^3 = \frac{\text{Power}}{\text{Power Density}} \quad (6)$$

Table IV lists values of "a" for various ratios of these limits.

Table IV Core Radii vs. Volume, Determined from Power Limits	
$\frac{\text{Limiting Power}}{\text{Limiting Power Density}}$ (liters)	a (cm)
1000	62
500	49
200	36
100	29
50	23

The smallest value expected might be about the last cited - 50 liters, corresponding to, say, 250 MW @ 5 MW/liter. At this radius, D₂O is about as good

as Be or BeO, and it improves relatively as radius increases. We are therefore on fairly safe grounds in specifying D₂O as the "canonical" reflector, with Be or BeO being equally acceptable for small cores (and, therefore, useful when the core coolant cannot be D₂O) and H₂O being best when the source emerges from a quite small volume.

D. Using the Core Flux

Under circumstances where the backgrounds can be made acceptable, it is desirable to get closer to the source to attain high flux. In fact, it is, in principle, possible to extract neutrons directly from the core. In this circumstance, the previous arguments are not valid. Instead, one must examine core criticality and power density more closely.

If the core power density is PD (MW/l), and the macroscopic fission cross section is Σ_f , then the flux is:

$$\phi = \frac{PD}{\Sigma_f} \times 3.25 \times 10^{13} \quad (7)$$

The macroscopic fission cross section is related to the critical mass and volume by:

$$\Sigma_f = \left(\frac{N_o \sigma_f}{A} \right) \left(\frac{M}{V} \right) \quad (8)$$

where N_o is Avagadro's number, σ_f is the macroscopic fission cross-section, A is the molar mass, M the critical mass, and V the core volume. Then,

$$\begin{aligned} \phi &= \frac{(3.25 \times 10^{13})A}{N_o \sigma_f} PD \left(\frac{V}{M} \right) \\ &= \frac{(3.25 \times 10^{13})A}{N_o \sigma_f} \frac{P}{M} \end{aligned} \quad (9)$$

where P is total power.

Thus, to get high core flux, we attempt to attain a highly dilute core (M/V small) or a small critical mass, depending on whether power density or power is limiting.

Additionally, we may get high epithermal fluxes by making the core undermoderated, and thereby achieving small values of σ_f . This factor is, however, counterbalanced by the fact that critical masses of intermediate reactors tend to be large.

E. Flux Traps

The ultimate in penetrating the core is to use the flux at its center. This poses no problem for lightly loaded reactors. However, heavily loaded small reactors, immersed in a moderator have peak core flux at the outer edge and peak thermal flux in the moderating reflector at about one rootage from the core.

This effect can also be observed in the center of the core. Geometrically, if one takes a slab core and deforms it into a cylindrical or spherical annulus, the thermal flux peak on both sides persists, and one of these peaks is now at the center of the annulus.

This peak is known as a flux trap, and has been well investigated.⁽³⁾ The most definitive study is that of Ergen.⁽⁴⁾ However, it is now well known that the performance of a flux trap depends strongly on core configuration.

The following heuristic points have emerged as a result of the many investigations of moderator islands in research reactor cores.⁽⁵⁾

1. There is a limiting ratio of core-to-moderator blackness, below which no flux peak in a moderator island is possible. This does not mean that experimental fluxes in moderator islands which are embedded in cores of low absorption are necessarily low; for such cores are characterized by already well thermalized spectra. The flux trap in such cores has degenerated to a simple filter removing virgin neutron background. Small quantities (dimensions small compared to thermal diffusion length of moderator) of materials such as D_2O or Be are optimal. In fact, maximum experimental flux per unit core power density can be obtained from such systems.

2. The flux trap principle peaks the thermal flux most dramatically when cores of high blackness to thermal neutrons are used. This means, simply, that in the core thermal flux is strongly depressed.

It follows from these two principles that the actual value of thermal flux in the center of a moderator island is only mildly dependent on core blackness, and, in fact is higher for cores of low blackness: the flux produced by thermalization of fast neutrons in the island is essentially additive to core flux.

The preceding argument assumes that "other things are equal." In practice, other things are not equal. For example, two idealizations of the moderator island assume that: a) the total source strength is constant (i.e., fast current at the edge of the island = $\frac{Q}{4\pi a^2}$, Q being available neutrons); b) fast current is constant at the edge. It turns out that geometrical factors prohibit flux peaking in the former case, which is appropriate to large, power-limited cores. Per unit source, the maximum flux is attained with a vanishing island. However, the source entering a small island is more nearly described by a constant entering current from a power-density-limited reactor (case b). Nevertheless, this current may be made somewhat larger by maximizing the inward leakage to the flux trap relative to outward leakage. To do this:

3. The core should be as thin as possible.

4. The core should be externally reflected by a good fast neutron reflector, particularly one with a high scattering cross-section for epithermal neutrons. Beryllium and nickel are recommended.

These two arguments can also be arrived at by simply examining the contribution to fast current entering the island from fission sources originating at various points in the core. The contribution is greatest for sources close to the island and can be somewhat enhanced by a good external reflector. The two points of view are therefore functionally related.

So far, we have not commented on steps to maximize flux within the island. The following points should be considered:

5. The "fast" neutrons leaking in should actually be as soft as possible.

6. The absorption cross-section of the flux-trap medium should be low, and the diffusion coefficient for thermal neutrons in it should also be low, to permit high flux buildup and inhibit thermal migration from trap to core.

All of these criteria are difficult to satisfy at once. For example, H_2O , D_2O , and Be can all function as flux trap media of approximately equal efficiency, but the dimensions of the optimum trap vary greatly (spherical holes of the order of 10, 70, and 35 cm radius, respectively), and the optima emphasize different properties of the media (the low age and thermal diffusion coefficient of water, the low absorption of D_2O , and intermediate properties of Be).

Although the flux trap is, ideally, completely surrounded by core, in practice, not too much harm is done by penetrating the core with a beam tube. As with irradiation experiments, not too many neutrons must be extracted via beam, or the flux will be depressed. In practice, about half of the neutrons in the trap may be used.

F. The Influence of the Experiment

In considering technological irradiations, the available experimental flux is often far less than the expected, due to the depressing effect of the experiment on the flux in its region. This same effect is apparent in beam experiments when the beam ports are large: a hole in the reflector is equivalent to an absorber.

It is necessary to consider the influence of beams in a research reactor in different ways, depending on whether the beam is large or small, and also depending on whether the beam port is essentially isolated or is one of a "lattice" of ports.

For a small isolated beam, the perturbation of the experimental source is small, and the flux is close to what is calculated without the beam. One word of caution is, however, necessary. Only half of the flux is directed outward, and that half is almost isotropic in space. Thus, the beam current available from a hole of length " l " and diameter " d " is approximately $\frac{1}{16} \left(\frac{d}{l}\right)^2 \phi_{\text{source}}$. For example, a 6" (15 cm) diameter hole, 3 m long, whose front end is in a zone of 10^{15} flux, has a beam current of $\sim 1.5 \times 10^{11}$ n/cm²-sec at the far end of the hole, and a beam strength of $\sim 1.1 \times 10^{13}$ n/sec.

As was mentioned, large beams depress the flux. Up to some reasonable size

this effect is not noticed by the experimenter, because the decrease in flux is balanced by an increased forward directionality of the neutrons leaking. However, the beam strength is more directly calculated from the neutron current than from the flux. Thus, a large "pipe" originating in a Be zone 40 cm from core center, which zone has normally a flux of 10^{15} , actually transmits a neutron current, $J = D_{Th} \left[\frac{1}{a} + \frac{1}{L} \right] \phi_{Th}(a)$, which is only $0.48[0.025 + 0.05] \times 10^{15}$, or 3.6×10^{13} n/cm²-sec. On the other hand, its tendency to forward orientation (cosine flux) yields $\frac{d^2}{4\ell^2}$ neutrons at the experimental facility, rather than $\frac{d^2}{16\ell^2}$, so that our previous case would have a beam current of 2.25×10^{10} n/cm²-sec if the neutron-current rule was taken.

The two rules - flux and current estimation - yield estimates of beam strength which vary by about a ratio of $4 D_{Th}/L$. This ratio is about 0.2 for H₂O and 0.037 for D₂O, which are in this circumstance the extreme cases. This uncertainty of a factor of 5-30 in the flux to be expected from an experiment is clearly too large for comfort. To do better, one must use more careful calculations or rely on experience. The only further rule of thumb is that small beams follow the flux rule, large ones the current rule, and that the measure of size is the diffusion length of the medium.

When one has a number of small holes, the flux should be computed as though the outer reflector were of reduced density. Then the flux rule may be used.

The current rule is, of course, pessimistic; for what occurs in the presence of a large hole is that the leakage down the hole is greater than the net current would indicate, and the outward current is reduced in the intact moderator. In essence, the beam steals neutrons from its environs. This may be augmented by leading neutrons to the beam face, a technique accomplished by enhancing thermalization rate at that point (as by using a hydrogenous material in place of another moderator). This technique is called "external flux trapping," for obvious reasons.

G. Core Life

Very high flux research reactors will operate at quite high powers - 100

to 1,000 MW, and maybe even more. At these powers, fissile fuel is destroyed quite rapidly. At 100 MW, about 1 kg is destroyed in 10 days, and at 1,000 MW, in one day.

For fluid-fuel reactors, the problems of burnup are solvable by straightforward engineering. For fixed-fuel reactors, the rapid attrition of the fuel has a profound effect on reactor design, both because it is desirable to have operating time long compared to refueling time, and because of the changes in core characteristics due to having a time variation in fuel and poison concentrations.

Conventional techniques (as used at MTR, ETR, and CP-5 for example) embody a refueling operation, performed after the reactor has been shut down and permitted to cool off, in which the fuel elements are removed and replaced, gradually and one at a time. The reloading shutdown takes two days to a week, depending on the cooling time adopted, the mechanisms available, the safety philosophy assumed and safety restrictions adopted. Thus, cycle times of 20 to 70 days are necessary to achieve 90% operating time.

Even with the shorter time - 20 days - and a 100 MW power, 2 kg of fuel are burned. Since thermal reactors are characterized by critical masses of 1 - 5 kg, this means that something of the order of an extra critical mass must be provided initially.

(Actually it is desirable economically to provide initially two or three extra critical masses; this permits amortization of core fabrication over a longer period.)

The large overload of fuel requires that there be an elaborate control system. The fresh core must be held down, either by burnable or movable poison, or by controlling leakage. Since leakage neutrons are what the experimenter wants, the latter method should be preferred, especially since core poisons additionally restrict neutron availability per unit power. Unfortunately, reactor designers have so far chosen the "conventional" approach of control rods rather than the "unconventional" one of reflector motion - to the experi-

menter's disadvantage.

To fulfill the requirements of core life, several expedients are, fortunately, available. In addition to the use of fluid fuel, which has several obvious advantages for the long run, the following points may be capitalized:

1. Intermediate and, particularly, fast reactor cores show relatively less reactivity saving over a large fractional core burnup. This both makes control generally easier, and makes it more feasible to adopt leakage control. Moreover, such reactors do not show large Xe buildup effects when shut down briefly.

2. There is no physical reason for not having brief shutdowns for partial refueling, or, in the limit, designing for on-line refueling. The required engineering development to permit this, and to satisfy appropriate safety requirements must ultimately be done.

H. Criticality and Power Shape

Up to this point, we have assumed that the high flux reactor core can be made critical with an assumed volume and reflector material and configuration, and with a low parasitic core absorption.

This is not always the case. Nearly all reactor types of potential utility can be made critical in volumes of 100 liters or more. However, attempts to capitalize on geometric factors by going to smaller cores require a progressive restriction of reactor type, and the smallest cores must be either proton-moderated or fast - a fact which is well illustrated in the choice of concepts for development as space nuclear power reactors.

Small cores pose another problem. They tend to be relatively black to low energy neutrons. Therefore, when immersed in a pure moderator, they see a large reentrant low-energy flux, which creates a power spike at the interface. While this is desirable experimentally - it produces neutrons close to the experiment - it is undesirable from the viewpoint of the core designer on three counts: first, the power spike is a region of unusually high burnup, and thus is unreli-

able over life; second, it limits total system power, because the maximum power density is at the spike; and lastly, the spike is likely to consist largely of epithermal fissions, which energy range is least productive of experimental neutrons.

III. POWER AND POWER DENSITY

Since it is, in principle, possible to get very large fluxes in very large reactors at quite low power densities but very high power,⁽⁴⁾ it is clear that one of the limitations of high flux reactors is economic: how much can we afford to invest in high flux capability?

A. Cost Limitations on Total Power

Without attempting to be definitive, one may examine the order of magnitude of costs involved. For this purpose, one may look, first of all, at the capital costs of reactors. There are a large number of studies available on power reactors, and some firm numbers have been quoted. However, rather than cite references, I shall present a generalized argument.

Nuclear power in the U. S. is now selling at about 5 mills/kWh in large size plants (~ 750 MW(e)).

About 2 mills of this is fuel cycle cost.

Of the remainder, about 1/4 is generator and electrical.

Thus 2-1/4 mills/kWh represents capital charges on the reactor system.

The total plant capital charge is then 2.25×750 \$/hr or 40,500 \$/day or 14.8×10^6 \$/yr.

Capital charges run about 12%/yr, so that the reactor system cost would be $\sim \$125,000,000$.

Plant efficiency runs around 30%, so that the thermal power would be around 2500 MW.

Assuming that cost follows capacity by a square root law, we would then estimate that:

$$\text{Cost} \doteq 78 \times 10^6 \left(\frac{P}{1000} \right)^{1/2} \text{ dollars} \quad (10)$$

where P is in MW. (Note that, on this basis, a reactor of 100 MW would cost twenty-five million dollars, a number which is not off scale.)

One could also guess at what experimental facilities might cost, and how much capital costs of large power reactors might be reduced when thermodynamic efficiency is not needed. These two factors tend to balance. The fact is clear, however, on these general grounds, that a 1,000 MW research reactor, equipped, would cost something close to 10^8 dollars. Since one hundred million dollars and one thousand megawatts are round numbers; and since some experimental systems in the physical sciences have been approved at costs beyond this; I shall assume hereafter that 1,000 MW is possible for the next generation.

At 1,000 MW, fuel burnup charges would be about \$12,500/day, or about 4.6×10^6 \$/yr. Assuming these to be, roughly, one third the operating costs, the reactor would cost something more than $\$1.5 \times 10^7$ /yr to operate, again a not inconsistent number.

Research program costs are, of course, not included.

To summarize this section: a number of primitive correlations and scaling laws have been evoked to justify a power of 1000 MW as about the maximum that could be attained within current cost guidelines.

B. Power Density - General

While total power depends primarily on the expenditures justifiable in constructing a research reactor, power density depends primarily on the state of the art of heat transfer engineering employed.

Three basic mechanisms may be used to remove reactor heat:

1. The heat is retained in the fuel matrix, which is physically transported out of the reactor; the heat is then transferred to the secondary system.
2. The heat is transferred to a single phase fluid coolant, which leaves the core and then transfers heat to the secondary.
3. The heat is transferred to a boiling liquid, which is condensed and subcooled outside the core.

It is possible to combine these techniques - heating, boiling, and superheating a substance in multiple passes, or both transferring fluid fuel and transferring heat from it to an in-core coolant; but all three cases pose formidable engineering problems.

As was explained previously, power density and power set the scale of the reactor system; and from an experimental point of view, for a given power, the smaller the core, the better. Thus, even at high powers, high power density is desirable.

In the Introduction to this paper, it was pointed out that current advanced research reactor designs feature power densities of 1-3 MW/l; and that the most rigorous engineering development might perhaps lead to a factor of 2 or 3 improvement. Thus, 10 MW/l represents a target end point for these systems - water cooled reactors - and, at the same time indicates the performance which would be desired for a more powerful cooling technology.

C. Flow Rates and Their Effects on Power Density

The homogeneous reactor is in many ways the simplest, and may be examined first. The basic fact is that a homogeneous system has a limiting temperature, beyond which either the chemical problems of containing the fluid, or the difficulty of maintaining its integrity as a medium, become practically insuperable. The three most likely homogeneous systems are aqueous, molten salt, and liquid metal (a sodium slurry or a molten U or Pu eutectic). Table V lists the pertinent approximate physical data, and, from these the value of G (mass flow rate) required to achieve 1000 MW in a core of ~ 100 l volume in the form of a right cylinder 50 cm in height and diameter. The formula is:

$$G = \frac{240 P}{C_p \Delta T} = \frac{240,000}{C_p \Delta T} \quad (11)$$

It may be seen that the liquid-heavy metal concept is inapplicable; if a fast homogeneous reactor is wanted, it must be in the form of a lighter metal slurry. Of the other concepts, molten salt seems best, although one might argue about permissible temperature rise for a long time.

The flows in homogeneous reactors illustrate the minimum flow required of heterogeneous reactors with the respective coolants. However, heterogeneous flow rates are typically somewhat higher. The basic reason is that, due to the fact that heat must be transferred, local temperature gradients exist in the heat transfer passages. A secondary reason is that, often, temperatures which are acceptable to the coolant are unacceptable to the reactor structure. Thus, flow rates of several thousand kg/sec are needed for typical 1000 MW heterogeneous reactors.

It has often been suggested that the inclusion of the latent heat of boiling in the heat-acceptance mechanism of the coolant will permit smaller flow rates. This is true, for example, with boiling water power reactors. However, this could not reduce flows by more than a factor of two in research reactors, where one starts with highly subcooled coolant, and the advantage disappears as the upper temperature limit approaches the critical point.

D. Problems Arising from High Flow Rates, and Others

The pumping rates just cited are very large, but achievable by commercial practice. In reactor design, there are other limiting factors. The most common ones are:

1. For homogeneous reactors it is necessary to lead the fluid into and out of its critical volume. This means that the inlet pipes must be appreciably smaller than the reactor. For the illustrative homogeneous water reactor cited in Table V, the 1100 kg/sec of flow, if confined in a pipe of 15.8 cm diameter (10% of core area) would have a linear flow velocity of 56 m/sec. This is somewhat beyond (by a factor of two or three) standard practice. The pipe would have to be remarkably strong and fatigue resistant; pumping losses would be very high because of pressure drops in the pipe; and so on.

Nevertheless, this approach is more likely to yield to development than the similar problem in heterogeneous reactors: The greater difficulty lies in the combination of the mechanical problems just described with the problems of resistance of the primary system components to corrosion. All homogeneous

reactor concepts feature fluids which are a good deal more corrosive than hot water; the corrosion is particularly aggravated in reactor outlet lines.

Table V			
Flows in a 1000 MW, 100 μ Homogeneous Reactor			
Material	C_p (cal/g-°C)	Assumed Permitted Temperature Rise (°C)	G (kg/sec)
Molten Salt	0.5	600	800
H ₂ O	1.1	200	1,100
Na	0.32	400	1,900
Pu	0.033	400	18,000

Finally, one of the features of a homogeneous reactor is a "core pot." The nuclear design often restricts this material to one of low absorption, whereas mechanical and chemical considerations often recommend neutron absorbers for this function.

2. For heterogeneous reactors, essentially the whole core area is available for coolant delivery, the coolant is generally easy to contain, and no core pot is necessary. However, once the coolant enters the core, it is split into very small channel flows, in order to maintain good heat transfer surface. Thus, very large pressure drops (and dynamic pressures) characterize any high flux heterogeneous reactor core. The limits are found in problems of flow distribution, fuel element anchoring and resistance to deformation, and the need for pressurizing the whole system just to accommodate the core pressure drop.

E. Other Approaches to High Power-Density

Besides attempting to run water through reactor cores even faster, or developing better primary system components for homogeneous reactors, several other techniques are possible, at least conceptually, for going beyond 10 MW/ μ .

1. The nuclear rocket program has pioneered the development of refractory fuel elements. With such fuel, a very high driving temperature difference

relative to the coolant may be achieved. This large ΔT makes it possible to consider hydrogen and helium as coolant fluids. These, in turn, have much lower viscosities, relative to their heat capacities, than water. The net effect is to ease the mechanical constraints on heterogeneous reactor coolant flow.

2. Even with water cooling, if the chemical problems of handling steam can be transcended, refractory elements could in principle permit total evaporation and superheating to occur without requiring excessive system pressure.

3. One further possibility is suggested by the developments in ablation cooling of reentry space vehicles; in a reactor, it may be possible to turn the heat transfer scheme inside out, so that ablation cooling can be used to remove heat from a hot fissioning gas.

4. Radiation cooling represents an ultimate limit and a new technology; the familiar tungsten filament in a light bulb operates at about 100 MW/l of metal.

Of course, a light bulb operates in the open; in a normal heterogeneous reactor core, radiation would merely augment convective heat transfer, provided that the coolant were opaque.

However, if we are permitted to consider plasma cores, we can get high enough temperatures so that the surface temperature supports an enormous power density. In general, for a spherical black body radiating at temperature T , the mean power density in MW/l is $1.72 \times 10^{-14} T^4/R$ where R is the radius (cm) and T is the absolute temperature. At $T = 20,000^\circ\text{K}$, a 50 cm sphere has a power density of 50 MW/l. At $150,000^\circ\text{K}$, the power density is 2,000 MW/l. Beyond these temperatures nuclear collisions excite inelastic levels and hence the purely fission nature of the process is lost. Increasing the power density by decreasing R calls for plasma densities above 10^{18} atoms/cm³; these are probably too high to be reasonable.

Containment schemes for such plasmas have been proposed and used as the basis of reactor concepts.⁽⁶⁾

5. Certain types of reactors - annular cores in D₂O, and cavity reactors

whose fuel lines the outside of the cavity - have such thin cores that it may be possible to achieve very high power densities by squeezing coolant through the shell, or by spraying coolant against it. In either case, the usual pressure-drop limitations no longer are binding.

6. In general - the comment above represents an extreme case - it is easier to achieve high power density in small cores, simply because flow lengths are shorter.

IV. SOME SPECIFIC SYSTEMS

A. Small Core Water Reactors

The most compact reactor systems which can be made critical are the proton-moderated thermal systems and fast reactors. In view of our familiarity through experience with them, water-moderated-and-cooled reactors must always be considered first.

There exists a body of information on the critical properties of these systems⁽⁷⁾ with H₂O reflectors. As practical solutions, with fission products, U²³⁶, etc., water-reflected reactors can be made to go critical in a variety of sizes. Cores from 20 to 100 ℓ in volume all would require less than 3 kg of U²³⁵, and would be quite thermal in their power spectra. A small core might have a 20 ℓ volume, a k_{∞} of 1.68, and a concentration of 100 g (U²³⁵)/ℓ; a middle sized one would have V = 50 ℓ, k_{∞} = 1.40, and c = 36 g/ℓ; a large one, V = 100 ℓ, k_{∞} = 1.25, c = 25 g/ℓ.

These cores would all be power-density limited. The figure of merit, Power × availability factor × geometry factor, can be approximated by: $\left(\frac{\text{Volume}}{(r+6)^2}\right)\left(\frac{k_{\infty}-1}{k_{\infty}}\right)$. This is, for the three sizes, almost identical.

The substitution of Be for H₂O as reflector would reduce critical core radii by 6-7 cm. The result would be to permit a critical volume of ~ 20 ℓ at a k_{∞} of ~ 1.39, and a volume of ~ 7 ℓ at k_{∞} ~ 1.7.

At a critical volume of 20 ℓ, a concentration ~ 35 g/liter and a power density of 10 MW/ℓ, power would be ~ 200 MW. The reactor would be similar to a cleaned up, more highly cooled MTR. With a 200 MW power, ν = 2.43, k_{∞} = 1.39,

approximately 0.045×10^{16} neutrons/cm²-sec represent current at experimental location, and approximate flux would be around 6×10^{15} n/cm²-sec.

(This may be compared with A²R², in which a somewhat larger core, at 100 MW, with a similar k_{∞} and pronounced leakage into a flux trap, has beam-entrance fluxes of $1-2 \times 10^{15}$; remember that my "flux" is a factor of two too high.)

Without detailed calculation, but from HFIR and A²R² examples, an internal thermal column flux of (truly) 2×10^{16} n/cm² would appear reasonable.

The burnout life of such a core would be very short - 5 days if an extra kg of U were loaded in - and, as heterogeneous reactor, a quick core turnaround would be desirable.

If no flux trap is wanted, such a system could be built also as an aqueous homogeneous reactor. The concentration of ~ 0.2 M uranium salts is acceptable, provided that system temperatures are kept below $\sim 250^{\circ}\text{C}$. Uranium salt solutions are generally acidic, and the usual corrosion problems of containers must be faced.

The smaller size, Be reflected system may be profitably examined. For one thing, high power densities are more readily attained in small sizes than in large. Moreover, more concentrated systems can tolerate structural neutron poisons more readily than dilute ones.

Although I have computed 7 ℓ as critical volume, reduced water density at power makes 10 ℓ a safer volume for a reactor of $K_{\infty} = 1.7$, $c = 100$ g/ ℓ . At 10 MW/ ℓ , this 100 MW reactor can be operated with about 1/2 the coolant velocity of its 1000 MW cousins. Its beam performance is virtually identical to that of the 20 ℓ , 200 MW system.

The problem of constructing a small, water cooled and moderated research reactor leads to an inevitable compromise: reduced size and power yield a reactor of good performance, but at fuel concentrations that are quite large. Although high specific power is easier to achieve, the concentrated core introduces large flux peaks. In the heterogeneous reactor, increased fuel concentration leads to decreased water density, and the smallest sizes are even more difficult to make critical; in the aqueous homogeneous reactor, the high fuel

concentration introduces increasingly difficult chemical problems. Backing off from the smallest critical reactor, which is optimal in terms of total power and heat transfer engineering, leads to HFIR, A²R², and other current-generation designs at various compromise points.

B. Heavy-Water Reactors

A crucial difficulty with light-water reactors is that, unless they are quite small, the critical mass of fuel must compete with a significant quantity of water absorber for neutrons. Systems of more relaxed size - 100 ℓ or more - have a low k_{∞} , and thus a small availability of excess neutrons per unit power.

The A²R² solution - going to a heavy uranium loading - to a first approximation replaces light water absorption with useless U²³⁵ absorption in an intermediate spectrum; its major virtue is partial alleviation of the loading-life problem, and a tolerance for neutron-absorbing structural materials, such as iron.

If one wants to keep the bulk of the fissions in the thermal range and yet have a large volume core, the obvious step is the replacement of light water by very weakly absorbing heavy water.

A practical heavy-water core, with some fission products and Xe, could go critical at 100 ℓ volume with a k_{∞} of 1.88 and a concentration of about 6 g/liter of U²³⁵.⁽⁸⁾ This is as estimated for a D₂O reflector, which is a convenient experimental medium. If, in some way, 1000 MW could be extracted, the mean thermal flux would be greater than 4×10^{16} , and the peak would approach 10^{17} !

The core would, as a sphere, have a radius of slightly under 30 cm. At an experimental point 40 cm from the core center, the current would be about 1.8×10^{15} n/cm²-sec, and the small-beam flux close to 6×10^{16} .

The great improvement over the light-water system comes from several factors:

1. The artificial one of this paper. Had I chosen 200 MW @ 10 MW/ ℓ , or 100 MW @ 5 MW/ ℓ , 20 ℓ would be the optimum size core. Although D₂O systems can be made critical at this size, their concentration approaches that of H₂O systems.

Residual advantages would be the factors below.

2. The higher k_{∞} of D_2O cores, making them more productive of experimental neutrons for a given power.

3. The higher flux/current ratio of D_2O reflectors, which again becomes apparent only in large sizes.

In summary to this point: D_2O systems with core sizes up to 100 ℓ have intrinsic neutronic advantages over H_2O reactors. If 10 MW/ ℓ core power can be attained, they are very attractive.

The low critical concentration - cited as 6 g/ ℓ - makes it feasible to reconsider solution cores for this concept.

The very high core fluxes also make it attractive to consider locating experiments in an unfueled central zone. The peak core thermal flux for a constant-power reactor occurs with the core of minimum critical mass. According to my estimate, this would occur with a core only slightly larger than the one just postulated - 120 ℓ , at which size the critical concentration would be 5 g/ ℓ . A 10^{17} flux would be attainable in a central D_2O tube in such a reactor.

This is not yet a flux trap. In D_2O , it takes a very "grey" annular core and a large gap to develop maximum flux.

The flux in a homogeneous reactor can, of course be improved further by on-site chemical processing to remove fission products; with a consequent decrease in critical mass. This can be done for this reactor. There is only one other engineering problem which was not mentioned in the discussion of obtaining high power density from homogeneous reactors: this is the problem that all structural parts, including core pot and entrance pipes, must be very weak neutron absorbers. Otherwise, the virtues of the low absorption in D_2O are seriously compromised. Materials such as Be, BeO, C, and Mg, with Zr as a possible cladding material, must be used.

As a possible alleviation of the high pipe-flow rates, an "octopus" or "dodecapus" might be used. It is sketched in Fig. I. Such a reactor will have neutron production to some extent in the arms, and I can think of no easy way

to figure out its real critical properties.

C. Small Fast Reactors

Concentrated fast reactors are the smallest that can be made. For example, the critical mass of a U^{235} sphere in a tub of H_2O is only about 23 kg⁽⁹⁾ corresponding to a core volume of 1.225 liter and a core radius of 6.64 cm. In D_2O , the critical radius is about 5.6 cm. Even a bare sphere (Godiva) has a radius of only 8.4 cm.

Plutonium spheres are even smaller, and a high density Pu sphere in thick Be reflector is cited as having a critical radius of under 3.2 cm.

These systems thus approach the "point source," and the possibility of capitalizing on geometry thus again must be examined.

Selecting a point at radius 15 cm in a Be reflector as experimental location, and assuming that a Pu source has $\nu = 2.9$, $k_{\infty} = 2$ (to account for the rather large fraction of epithermal fission), we have a net neutron availability of 4.7×10^{16} neutrons/MW and, at the experiment, a neutron current of 1.66×10^{13} /MW and a thermal flux of almost 3×10^{14} /MW.

A conservative example of the virtue of this system is illustrated below. It is conservative because performance could be markedly improved through use of plutonium and incorporation of very primitive internal cooling.

Let us consider a concentrated uranium sphere immersed in a 5 cm thick sphere of H_2O . The critical radius would be under 7 cm⁽⁹⁾ and the total assembly radius would be 12 cm. The thermal conductivity of uranium is good enough so that this assembly could operate at 8 kW of power for a uranium temperature rise of 200°C. At this power, given $k_{\infty} = 2.2$, the beam current from the water surface would be 1.9×10^{11} n/cm²-sec. In a graphite hohlraum, this current may be somewhat intensified. This beam current is about what one would get from a 10^{12} flux research reactor; admittedly, not a high flux system, but extremely good performance when compared with a 100 kW swimming pool!

When such a reactor is opened up for cooling, it can still be quite small.

We consider now the next largest size, that of EBR-I⁽¹⁰⁾ and Clementine.⁽¹¹⁾

Experimental Breeder Reactor-I, as an example, has a core a little over 6 ℓ in volume, corresponding to a sphere of about 11 cm radius. With a most conservative cooling system, it produced 1 MW of power. Its k was about 2.1.

We now know how to achieve power densities of 1 MW/ ℓ in large sodium-cooled fast reactors, and it would be straightforward engineering to do the same for a 6 ℓ fast reactor. The use of Be, instead of U, for blanket would markedly increase reactivity, making it possible to use smaller rods and more sodium in the core.

A higher power density than 1 MW/ ℓ may be predicted for this type of system. Sodium coolant flow of 15 m/sec seems possible, and at 30% core porosity and 10 cm core radius, this would amount to a mass flow rate of 115 kg/sec. At a heat capacity of 1.25 joules/g- $^{\circ}$ C, and a temperature rise of 400 $^{\circ}$ C, this would permit a power of 57.5 MW, or almost 10 MW/ ℓ .

The key to this high power density is the small core size, which permits the coolant to have a very short core residence time. Of course, all the numbers cited are advances over current practice (EBR-II, for example, has a core 2.5 times as long, sodium velocity 1/2 as great, and temperature rise 1/3 to 1/4 as much, in achieving somewhat less than 1 MW/ ℓ). Nevertheless, the sodium route is one of the more straightforward ways of achieving high power density.

At 60 MW in a 10 ℓ U²³⁵-fueled core with a Be reflector, we might achieve $k_{\infty} = 2.1$. With an experimental position at 24 cm, the experimental current be 3.4×10^{14} , and the small beam flux, around 7×10^{15} .

A somewhat larger core could undoubtedly be built of refractory materials. We have investigated W-UO₂ cermet systems at Argonne, for heat transfer nuclear rocket application. At about 50 ℓ and up, it becomes possible to extract 1000 thermal MW from them. The key is the use of hydrogen coolant, which can achieve high flow rates in small passages without unusual pressure drop, and has a very high heat capacity per unit weight. The refractory core permits the heat source to sustain a high driving temperature difference to the bulk coolant, for heat

transfer.

Such a system - 1000 MW @ 20 cm radius, with $k_{\infty} \sim 1.8$, would yield the impressive flux of $\sim 1.6 \times 10^{17}$. Unfortunately, it is probably too costly to supply many cores of this type, and the cost of using hydrogen as a coolant is also extreme - as is likewise the hazard. Nevertheless, it indicates that technology in itself does not limit power density.

D. Solid-Moderated, Gas-Cooled Reactors

It has already been observed that high flux may be associated with reactors of large volume and relatively low specific power, or with gas-cooled reactors built of refractory materials. Although these are two distinct types, the favorite materials are identical: homogeneous BeO-UO₂ with coolant channels, or homogeneous C-UC₂, with coolant channels. The specified coolant is helium, which combines low viscosity, high heat capacity, and lack of chemical reactivity.

1. Large, Low-Power-Density Reactors

The "Treat" reactor⁽¹²⁾ is the prototype of a graphite-moderated-and-reflected thermal reactor. With a critical mass of 5.1 kg of U²³⁵ at a core volume of 1750 l, it is close to the minimum critical mass for practical graphite reactors.

This loading would yield a mean flux of $\sim 5 \times 10^{12}$ P (power in MW) and a central flux of $\sim 1.5 \times 10^{13}$ P. The central flux in such a large reactor is readily accessible, as TREAT experience has shown. The equivalent of a large beam has been burrowed into the center, in order to observe fuel-element-sample transients visually.

Thus a beam flux of 10^{16} is available from a TREAT-type reactor for a power of about 700 MW and a specific power of 400 kW/l.

Let this core be a right-circular cylindrical core of L/D - 1, whose height is 82 cm, and radius 41 cm. Let core porosity be 10%. Then coolant flow area is 525 cm². Assume that the coolant is helium at 100 atmospheres pressure, and that permissible temperature rise is 1500°C. Required mass flow is 88 kg/sec,

and inlet coolant velocity (assuming 273°C inlet) is 190 m/sec. Although all these numbers represent large engineering works, none is beyond feasibility.

Some confirmation is presented in the design of UHTREX.⁽¹³⁾ This reactor has a room-temperature critical mass of 3.75 kg, but a practical critical mass of 6.8 kg at temperature. Its core volume is about 2500 l. UHTREX is designed to operate at a power of only 3 MW, with a pressure of 34 atmospheres, a coolant flow rate of 1.3 kg/sec, and a coolant temperature rise of 450°C. The pressure drop is only 9 psi through the core. The designers believe that a factor of 10 in power is attainable merely by increasing blower size and heat rejection capacity. The next factor of 10-30 in power requires increased system pressure and a high-power compressor.

2. Moderate-Size Reactors

The scale of a reactor is set by the age of fission neutrons in the moderator selected. Thus, graphite systems have characteristic dimensions of a few twenty's of cm. The age in BeO is about the same as that in D₂O, and the sizes of optimal BeO moderated reactors are not much different. However, the absorption cross-section of Be is much larger than that of D₂O (although much less than that of H₂O).

The net result is that reflected BeO-UO₂ systems have minimum critical masses of under 5 kg of U⁽¹⁴⁾ at core volumes around 100 l. Presumably, the basic design features of UHTREX could be incorporated to permit rapid fuel change, so that not too much excess fuel would need to be included. As a high flux core, this is significantly inferior to a D₂O solution; however, it is readily adapted to a gas-cooled design. (See, for example, EBOR program concepts.⁽¹⁵⁾ The EBOR program has been aimed at civilian power reactors, with power densities of only tens of kilowatts per liter. However, it is clear from previous work on PLUTO⁽¹⁶⁾ that much higher power densities may be contemplated.)

E. Molten Salt Reactors

The basic information I have used is that presented in the nuclear analysis of MSRE.⁽¹⁷⁾ It was pointed out earlier that molten salts have, as homo-

geneous reactor vehicles, the best product of heat capacity and permissible temperature rise, so as to give relatively low mass flow rates per unit of power generated.

The MSRE is actually a heterogeneous reactor with fluid fuel. The core is large - 70 cm in radius. Power is 10 MW, and temperature rise is only about 30°C. Under these conditions, and for a fuel unburdened with fertile materials, peak core flux is 5.56×10^{13} . The k_{∞} for this system is about 1.5.

The fuel temperature rise reported is quite low, and it should be possible to achieve a factor of 10 increase, so that a power of 100 MW and a peak core flux of 6×10^{14} should be realizable. The system is too large for much external beam flux, but the use of a Be flux trap could probably achieve 2×10^{15} flux.

The core is mostly graphite moderator. As the graphite is removed, the system becomes more compact, but less of a thermal reactor. In the limit, it is possible to build a molten salt reactor as a Be-reflector-moderated system in a thin shell, for which Ergen⁽⁴⁾ computed a 7×10^{15} central-island flux @ 100 MW.

I do not have available a calculation of the properties of a molten salt reactor as a compact sphere. If the neutronic properties of this system are favorable, it should have comparable performance to the high powered EBR-I illustrated in Section IV-C. One reference⁽¹⁸⁾ cites figures indicative of a 1 g/cc fuel concentration, which is appreciably less than EBR-I and indicates criticality in a larger volume.

F. Cavity and Thin-Shell Reactors

It has been often observed, in this paper and elsewhere, that a major limit to power density in heterogeneous reactors lies in the relatively long coolant flow path through the core. Pressure drop increases with length, and temperature rise is constrained to some maximum. It is for these reasons that smaller cores may have higher power densities.

It is clear that shorter coolant path lengths may be achieved with radial coolant flow. For a more-or-less compact core, only a factor of two reduction in flow length is possible, and this is hardly worth the design difficulty.

However, there are core types for which the configuration of fuel can be a thin annular section. For such cores, quite short flow paths can be realized by radial flow.

The most intriguing of these systems are the D_2O flux trap cores, and their close relatives, D_2O cavity reactors. Several reactor systems of this general sort were investigated some years ago.⁽¹⁹⁾ The basic observation is that, in a large body of moderator (D_2O being the best example), a spherical shell of fuel just thick enough to be black to thermal neutrons will be critical.⁽⁴⁾ If the shell is made larger, its blackness can be reduced to some limiting value, which is that required for criticality in a thin slab. Moreover, if the interior of the shell is evacuated, only a slight increase in blackness keeps the system critical; and if the contents of the shell are dispensed uniformly within the sphere (the cavity reactor), again only a little more fuel is required.

The cavity reactor is often conceived in terms of a fissioning gas, with temperatures approaching those of plasmas. It is a possible core for plasma-reactor schemes, such as those of Ref. 6. The cavity reactor may also be fueled with a suspension of fissioning "dust;" as such, it is effectively a gas-phase homogeneous reactor of potential quite similar to that of the large homogeneous heavy-water solution. Problems of corrosion and plateout are replaced by problems of erosion and adhesion.

However, the existence of thin-shell critical assemblies presents a more novel reactor concept. Figure II illustrates a possible configuration: The core is a thin shell of a strong, fueled material - such as a stainless steel-urania cermet. This shell is perforated with a very large number of small holes, possibly drilled or punched at the time of fabrication, and probably clad by a chemical or electrochemical technique. The shell has at least one reentrant thimble for high flux irradiations and a stem for coolant entry. Coolant is pumped in, forced through the holes of the colander, and dumped into the main moderator vessel.

Although the inlet pipe must contain the high flows characteristic of homo-

geneous reactors, and the shell must support significant pressure, the high-temperature outlets of the homogeneous reactor have disappeared, and pressure drops do not approach those of parallel-flow heterogeneous systems. The D_2O may be permitted to evaporate and superheat in passing - it will recondense in the outer tank. (Not shown is the main cooling system, which withdraws warm moderator from the tank and returns cold by downward circulation around the outside of the shell.)

For those who are concerned with the strength of such a colander, an alternate scheme has been proposed at Argonne.⁽²⁰⁾ The reactor shell is now solid (Fig. III). The coolant is dispersed by flow through a nozzle head and impinges on the shell's inner surface. The velocity of the coolant jets is high enough so that they can cut through the steam which comes off that surface, impinge on it, and be evaporated. The outer surface of the shell would be vapor-bound in stable film boiling but not at a high enough temperature to burn out.

Finally, the scheme may be reduced to more conventional terms by reverting to parallel flow in small tubes of some appreciable length. Such a reactor looks like the colander of Fig. 1, with the thin shell replaced by a tubular fueled structure of 2-10 cm thickness (and flow length). This reactor has been sketched up and studied at Argonne.⁽²¹⁾

The thin shell may also be used in reactors of more normal flow design. D_2O flux traps may be simulated by heterogeneous designs consisting of individual fuel elements spaced around a cylindrical annulus or a pseudospherical "basket." The large radius of optimal D_2O flux traps permits, in fact, enough space for a pressure-tube reactor design, which might make on-line reloading more feasible.

F. Neutron Sources Manufactured in Production Reactors

From time to time, it has been suggested that the reactor analogue of accelerators for neutron production is the manufacture of radioactive sources. If these sources can be made small, they have advantages of size - the geometric factor of a small source, and the greater ease of extracting high power

density in small size at low power. In the absence of multiplication, all neutrons are available for experiment. Finally, the difficulties of reactor operation are segregated from those of operating a beam research facility, and, in fact, a proper reactor for source production already exists at Savannah River.

Thus, it is necessary to discuss this possibility.

Four isotopes which may be of interest are: Cm^{244} , Cm^{248} , Cf^{252} , and Cf^{254} . The first of these is primarily a (α -n) source; although it shows some spontaneous fission, the α /fission decay ratio permits more neutrons to be produced by the alphas on Be. The other three are spontaneous fission sources. All are manufactured from Pu^{242} , which is a material produced in power reactors and of little use therein; it may be considered a "free" source material, for although it is not very common it will ultimately be produced in quantity.

The production rates and abundances reported here derive from the Argonne heavy-element cross-section data.⁽²²⁾

1. Cm^{244} . The production of Cm^{244} from Pu^{242} involves the absorption of two neutrons, each followed by a short-life β decay. Conditions of exposure may be adjusted so that yields of 70% are probably practical. If it is assumed that the two neutrons came from fissions in Pu^{239} , the basic cost of one atom of Cm^{244} is the cost associated with Pu^{239} burnup to produce three excess neutrons. This comes to about \$20/g in burnup charges. (I cancel production reactor operating cost against research reactor amortization and operating cost.)

As an (α -n) source, a (Cm-Be) system could probably produce about one neutron per 10^4 disintegrations. The energy release per neutron is then about 60,000 MeV, or a factor of about 300 more than fission in a reactor (assuming that the reactor produces one excess neutron per fission). Thus, the Cm-Be source would clearly be non-competitive with even low-power fast or water-moderated reactors.

2. Cm^{248} . The production of Cm^{248} from Cm^{244} involves passing through Cm^{245} and Cm^{247} , both of which are fissile isotopes. As a result, it takes, at 80% yield efficiency, 36 atoms of Cm^{244} and essentially no neutrons, to make

one atom of Cm^{248} . The base cost of Cm^{248} is thus \$1080/g.

Cm^{248} has a long life and a high ratio of spontaneous fission to α -emission. It produces one neutron/2.83 decays, or about one neutron/68 MeV. It is therefore, on a heat basis, appreciably better than a chain reaction.

Unfortunately, its long half life results in a neutron production of only 4.4×10^6 n/sec-g. While a 1 kg quantity can ultimately be envisaged, this is not even a 10^{10} source. The radioactivity is too weak.

3. Cf^{252} . The production route from Cm^{248} to Cf^{252} passes through fissile Cf^{251} , with the result that, at best, only 0.4 atom of 82 results from 1 of 68. Assuming production losses, we might burn 3 atoms of (68) to actually get our product. Neutron charges are now trivial, but the base cost of Cf^{252} thus is about \$3,000/g.

Cf^{252} has an α /spontaneous fission ratio of 32, and produces about 3.7 n/fission. It therefore produces one neutron/103 MeV, about a factor of two better, on a heat basis, than a reactor.

Its half-life is 2.52 years, so that it has to be considered a consumable. A 10^7 \$/yr fuel allowance, at base rate, is a consumption of 3.3×10^3 g/yr, yielding a usable inventory of 12 kg. At a specific source strength of 2.33×10^6 n/sec- μg , this is a total source of 2.75×10^{16} n/sec, at a power of 0.45 MW. The flux from this taken as a 5 cm radius sphere (coolant included) in a tub of H_2O , with experimental point at 8.2 cm, would be 5.8×10^{14} n/cm²-sec.

The result here is tantalizing. If a way could be found to circumvent parasitic fissions, particularly in the production of Cm^{248} , the product Cf^{252} might be inexpensive enough to justify its use as a source for 5×10^{15} flux. The production power requirement would be high - 3600 MW, even at 1% yield from Cm ; but this might be worth the trouble.

4. Cf^{254} . Cf^{254} would be a superior source. It produces 3.7 n/fission with no competing α decay, or neutrons at 55 MW/n. Its 55 day decay rate yields a source strength of 1.19×10^{15} n/sec-g.

Unfortunately, its manufacture requires neutron capture of Cf^{252} (2.64 yr,

9b) and Cf^{253} (19 day, 2b). To get a 90% yield of 84 from 82 requires a production flux of 2×10^{18} , and even a 50% yield would require 1.2×10^{16} . These numbers assume thermal neutron capture. Resonance reactors might make the production of Cf^{254} more feasible, provided that the cross-section of Cf^{254} does not increase relative to its parents. A low yield would nevertheless be tolerable, since one decay of Cf^{254} produces as many neutrons as 33 decays of Cf^{252} .

V. SUMMARY AND OPINIONS

Present high flux reactor design has emphasized the light water-cooled-and-moderated reactor type which has evolved from the MTR. This reactor type is now approaching technological limits, primarily because, at sizes dictated by burnup life and neutron flux irregularities power density becomes limiting.

Overpowering power density limits in this reactor type would require a brute force development, and would probably not result in reactor improvement by an order of magnitude, except at an incommensurate cost. It seems more logical to attempt to sidestep the problems by going to another reactor type.

If the fissions are either fast or thermal, plutonium should be seriously considered as a research reactor fuel. It produces more surplus neutrons per unit power, at smaller critical loadings than U^{235} . An advantage not mentioned elsewhere is the utility of Pu^{240} as a burnable poison.

Reactors operating at powers up to 1000 MW might be justifiable at national research centers; nevertheless, the need seems to be more for high flux than for a very large number of facilities. This favors small reactors at high power density, since there is a significant geometrical factor, in their favor for high flux and against for numerous holes.

Limitations of core lifetime can be overcome by any one of three methods: use of fluid fuel; use of fast reactor cores, with leakage control; or design for on-line or very rapid refuelling. One or all of these methods must be used in the next generation.

Fluid flow limitations caused by inadequate temperature capabilities of

water coolant may be circumvented by going to larger channel fluid fuel systems, or to liquid metal or gas coolants.

The properties of heavy water are remarkable enough that, almost uniquely, one can resort to geometrical trickery to get high flux.

In consideration of these factors, I should like to state my opinions as to which type of design is likely for the next generation. I believe that one or more of the following three systems will be favored.

A) Compact fast reactors in beryllium reflector, with sodium cooling. Sodium technology and refractory fuel technology are both developing rapidly. In small fast reactors, very large coolant temperature rise at conventional high flows (10-15 m/sec) should be achievable. Power densities greater than 10 MW/l for cores of 10 l size are probably attainable to give beam fluxes of $\sim 5 \times 10^{15}$. Power spikes, while severe, can be dealt with by grading fuel-diluent ratios and providing appropriate coolant channel variation.

B) High flow D₂O homogeneous reactors. At 100 l size, these provide high flux in a central thimble and adequate beam flux; the critical mass is at a minimum, chemical refueling eliminates the burnup problem, and the critical concentration is low enough so that the severe chemical problems of HRE would be significantly reduced. Ten MW/l is a possibly attainable power density, and 5 MW/l should be straightforward.

C) Some form of D₂O flux trap. The penalties for an oversize flux trap are very modest in D₂O, making it possible to have a thin shell reactor of rather large core volume. One possibility is to approximate the shell by an annulus of pressure tubes; on-line or quick-change refueling is conceivable in such a design, and beam tubes could penetrate into the flux trap between the fuel tubes. Other possibilities are a fluid fuel annulus (molten salt?), or, assuming favorable results of the necessary engineering research, the colander or sprinkler designs.

While these systems all have excellent promise for outperforming the best current reactors by factors of about 5 - and, perhaps ultimately, 10 - they, too,

have foreseeable limits. It is likely that a flux greater than 2×10^{17} will never be achieved in a steady state research reactor. There is some possibility of "hiring" the Jackass Flats hydrogen pumping capability for an occasional hour. Under these conditions, and with appropriate reactor design, 50 MW/l and fluxes up to 5×10^{17} may be contemplated; but it would be a very expensive experimental run, indeed.

Since the containment of fissioning plasmas is likely to be only slightly easier than that of "fusioning" plasmas, I can speculate that the high power density of a radiating plasma, and the accompanying high neutron flux, may ultimately be achieved - in a thermonuclear device.

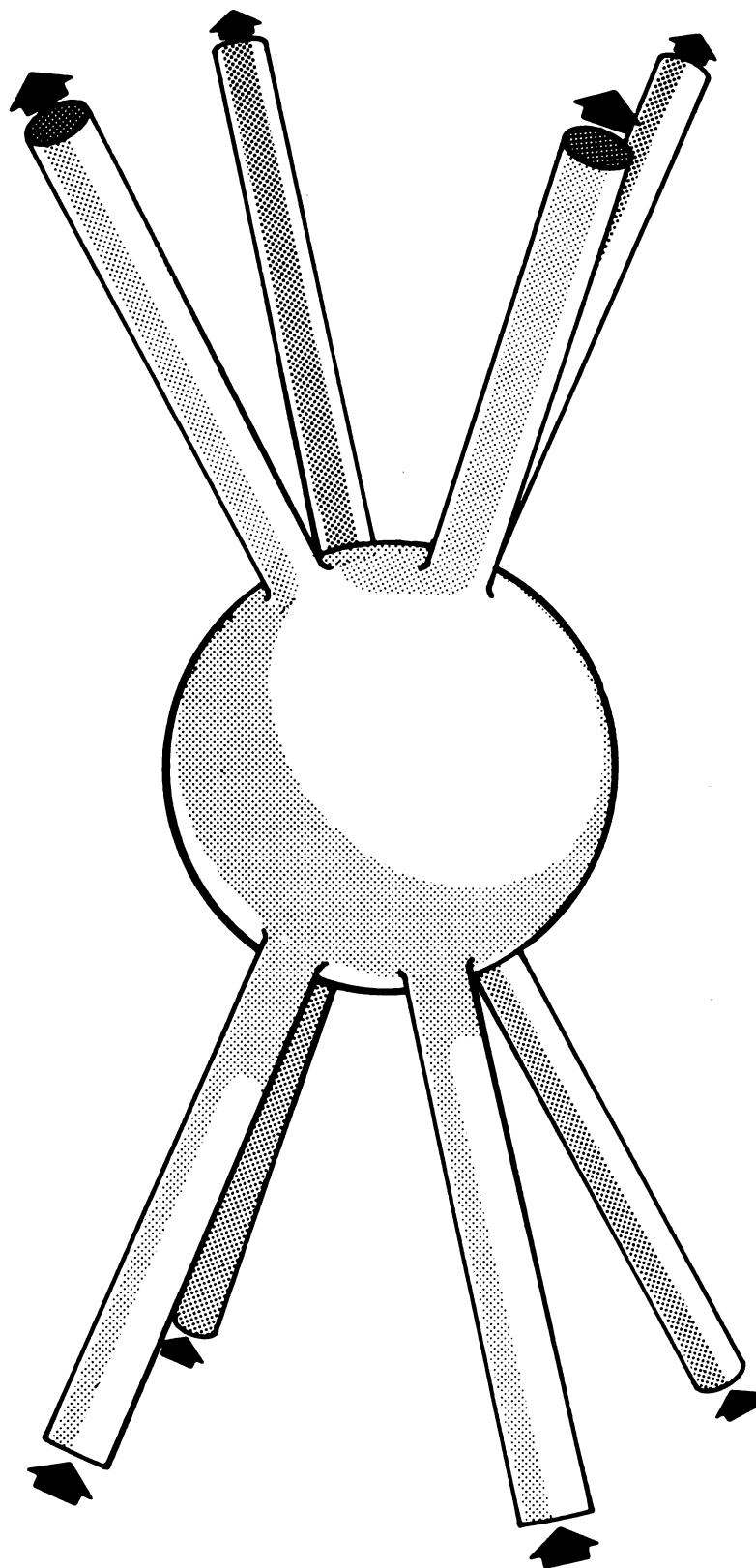


FIG. 1 AN "OCTOPUS" STYLE HOMOGENEOUS REACTOR, DISTRIBUTING
OUT-OF-CORE FLOW OVER FOUR LOOPS

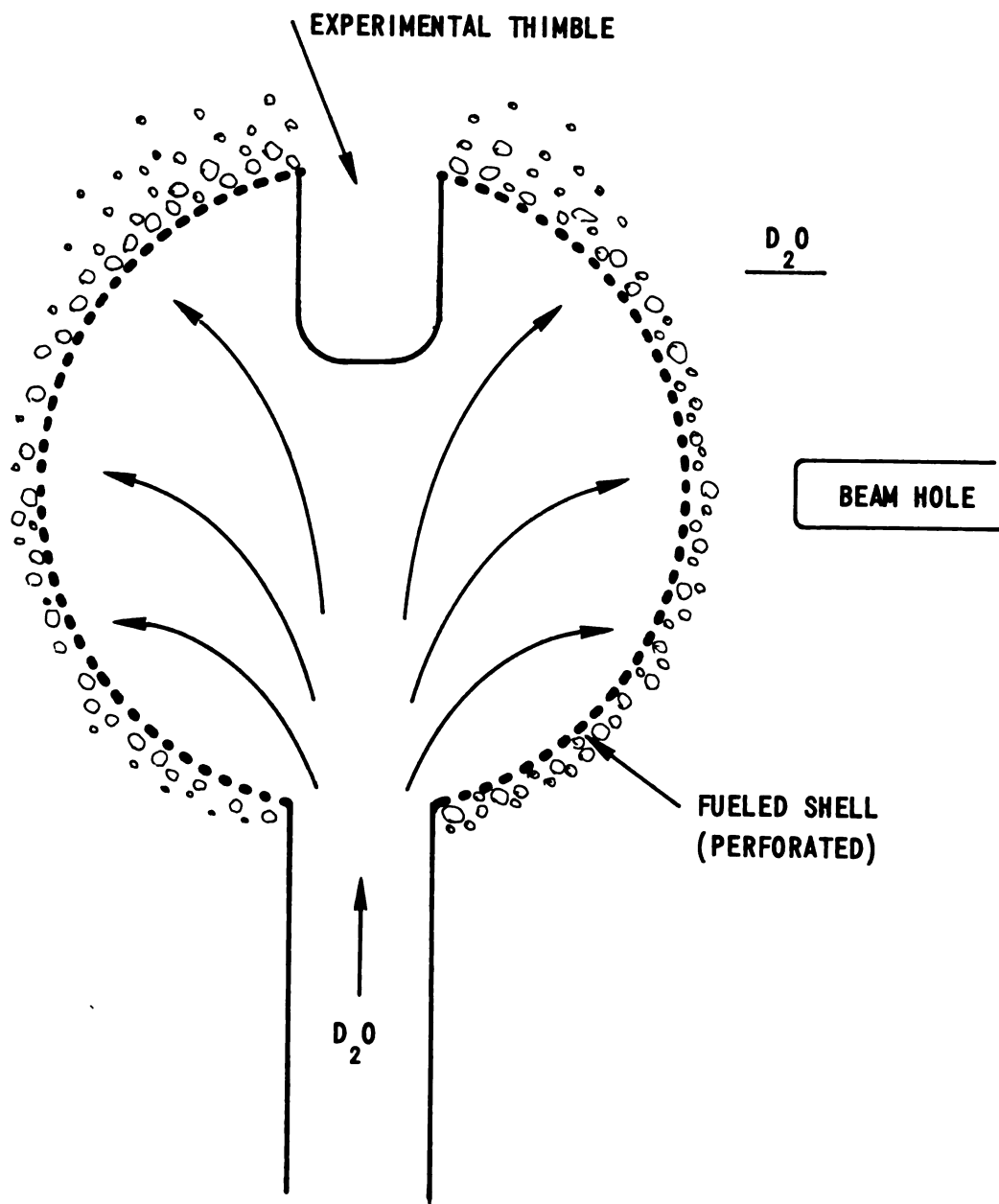


FIG. II A D_2O THIN-SHELLED REACTOR (COLANDER)

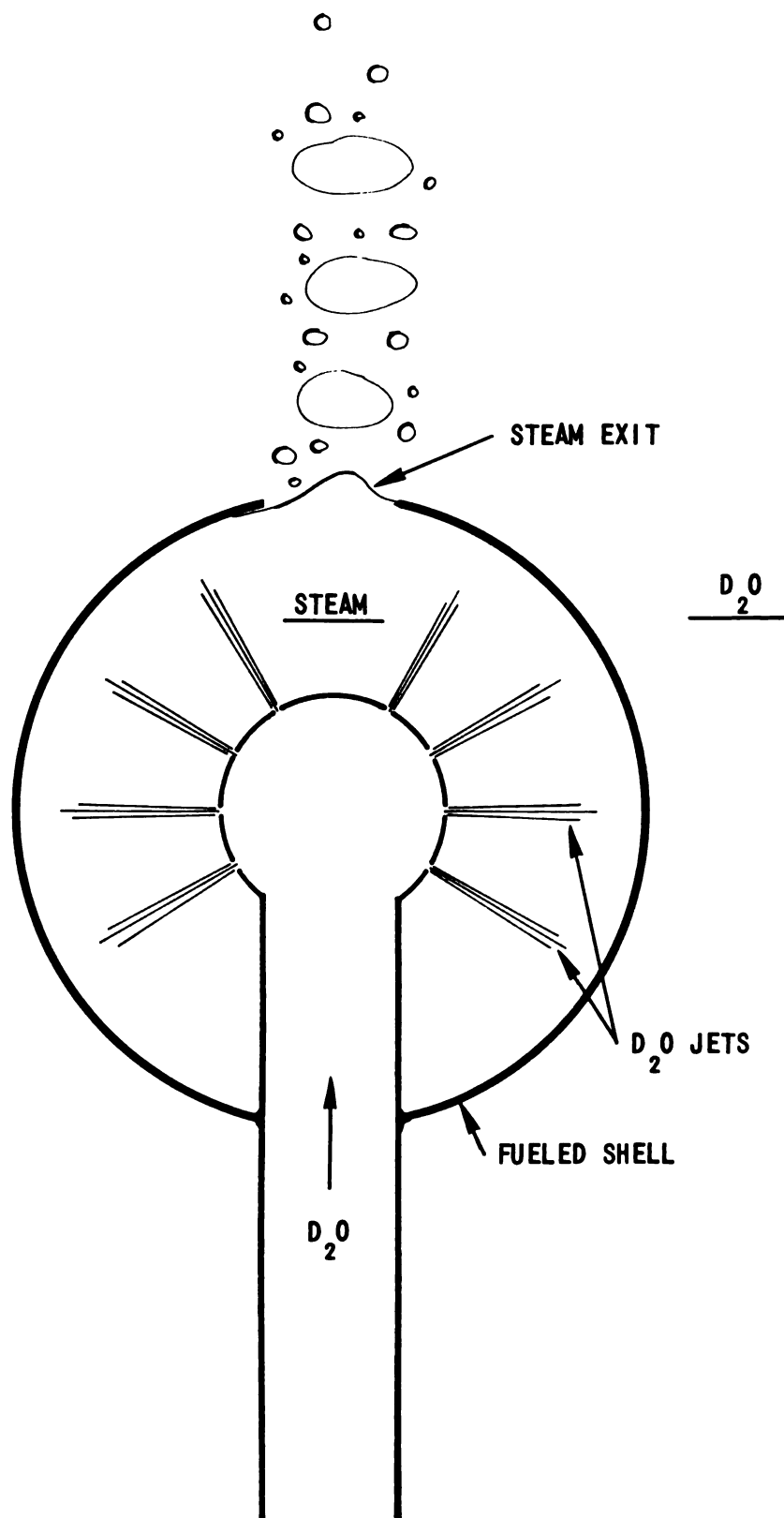


FIG. III A D_2O THIN SHELL REACTOR (SPRINKLER)

BIBLIOGRAPHY AND NOTES

1. The other papers in this Session are representative of this current generation of research reactors.
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-DISCUSSION
OF
PAPER II.B. (Spinrad)

Chairman: K. H. Beckurts

Secretary: C. Anderson

KEIBER: When Spinrad was writing this paper, I was doing some detailed calculations on the effect of replacing U^{235} with Pu^{239} atom for atom in something like an EBR-I core surrounded by heavy water. I found that I could, in fact, maintain the reactor at criticality while the core was surrounded by a completely black boundary so that all the neutrons from the Pu fuel core could be utilized in the experiment. None had to be reflected back into the core. This gave something like a 75% gain in the number of neutrons that could be utilized in experiments in this system, so that in this system (I investigated other systems too; this is by no means an atypical figure) the Pu fuel concept for research reactors represents a relatively easy way of getting a significant advantage at the present time with no great investment in technology development.

BECKURTS: Another point might be that in a couple of years the price of Pu might be lower than that of U^{235} . You are operating these reactors fairly close to burnout; doesn't the fact that the delayed neutron fraction is so much smaller in Pu present a potential hazard?

SPINRAD: I do not think that you can rightly claim that as an intrinsic hazard in itself. The fact is that any reactor that operates close to the margin must depend primarily upon some intrinsic prompt negative reactivity effect. In most of the research reactors the prompt reactivity effect is one of, basically, volume expansion of one sort or another. If you design properly for this volume expansion, it makes very little difference what the delayed neutron fraction is.

BECKURTS: Would you believe that operating research reactors on Pu might also be interesting in order to produce greater amounts of higher-mass Pu isotopes as an initial material for transplutonium production? Would one produce considerable amounts of Pu^{240} in that way?

SPINRAD: I mentioned in the paper that one of the bonuses one gets is when there is some 240 in the system (and it is a good burnable poison)-but it depends very much on the type of system. In a small core fast reactor one might be tempted to take some credit for taking bad Pu from the thermal power plant and making it all into fission products. That is the only way to do it. The use of research reactors as

higher isotope factories has something to commend it, of course, but one would require either the very large Savannah River size (~1000 MW) or a very good agreement among a large number of research reactors before you really made enough that way to be significant in what is going to be, I'm convinced, an enormous demand for these higher isotopes. I think that the demand will be so great that we will be forced ultimately into special production systems.

SMITH: If we have 1000 MW of low grade heat, I wonder if we can't take advantage of some of this heat. You are estimating 5 mils per kWh, \$30,000 operating cost per day, and these just about equate. Another opportunity is perhaps desalination; it might pay the whole bill.

FERRY: I am sorry Spinrad did not have the opportunity to pursue his remarks a little further about homogeneous reactors because I do think that they ought to be examined for this application. Unfortunately, there is some question about the power density that can be achieved in homogeneous reactors. In aqueous homogeneous reactors there is a problem of the radiolytic dissociation of the solution. I do not think that any of us can really say at the present time what will ultimately be the limiting power density. In any homogeneous reactor where there is an aqueous solution or a molten salt there is the problem of heat generation in stagnant layers near the boundaries of the container. I do not think that for molten salt reactors we really know how high a power density might ultimately be achieved, but it does not seem at all out of range to speculate that 10 MW per liter could be achieved. I would like to underscore a point that Spinrad made, i.e., if you try to go to higher total fluxes for a given limitation on the power density, the volume of the core must increase and the level of power density that can be achieved goes up much more slowly than the total power. So that to go from 10^{16} to 10^{17} , for example, will in general require much more than an order of magnitude increase in the total power. If one lays aside the obvious difficulties associated with building extremely large reactors (I don't mean technical difficulties, but economic difficulties primarily), one can visualize reactors which would be capable of steady-state operation at 10^{17} flux. In extrapolating the calculations that Ergen has made at Oak Ridge on molten salt reactors, one can visualize reactors which, for a few thousand MW of heat, could produce fluxes of 10^{17} n/cm²/sec. It is, of course, obvious that they would have very high operating costs unless, as Smith says, these costs could be partly recovered.

BECKURTS: I have two questions for Spinrad: When could such a fancy system (a 1000-MW homogeneous high flux reactor) operate? What kind of development effort would be involved in such systems?

SPINRAD: The problems that are already mentioned are quite clear. There is a tremendous radiolytic gas problem, and you have to understand this quite well before you can truly design. The other problem that I would consider quite outstanding is

the mechanical one of providing for changing the pot fairly frequently, because I would tend to think that any core pot for such a system as this could not be designed as permanent without being thick enough to seriously perturb the flux, and I would rather see that the system had mechanical features which involved occasional shut-down, quick removal of the pot, and reattachment of a new pot.

PERRY: I think the problem of corrosion which Spinrad alluded to earlier is, although certainly a problem, not the most serious one. I think in the case of molten salt, for example, if the temperatures of the molten salt are not in excess of 500° or 600°C, that the corrosion rates have been demonstrated to be really very low. I think that radiation damage problems to the materials are the serious ones, and this is true whether the salt in the high flux region is in contact with a material like graphite, which is a somewhat poor substitute neutronically for beryllium oxide but has also been shown to be very satisfactory in contact with molten salt at high power densities. Whether it is a graphite or a nickel alloy container, in either case I think that radiation damage to the containing material would be one of the key problems which would limit the life of the mechanical components of the system. Otherwise, I think that the technological developments are probably not great, pending satisfactory demonstration of the molten salt reactor experiment, for example. Again, the economical evaluations are probably one of the key obstacles at the present time.

BECKURTS: These high flux reactors would create new categories of problems for experiments. I think, for instance, to construct a cold source for such reactors would not be easy.

RAJEVSKI: Why are you so anxious to have such high power reactors? You ask the same question, because if you are going to such high power reactors, you have the problem of the cold source. I believe that after the discussion of the first paper it seemed that it is a very difficult problem with the power which is now achieved with the high flux reactors, so very likely it will not be possible to solve this problem for a reactor with 10 times more specific power. So you lose this advantage. It would be a reactor which must be specialized for certain types of experiments but it would not have the large application that you can have with high flux reactors in the range of 10^{15} flux.

BECKURTS: I think one point made by Spinrad which is very important was that these very high flux reactors must be such that they do not have to have a very high built-in reactivity for long fuel cycles, but that they either will have continuous refueling or on-line refueling. We have always had the experience that it is nice to get a reactor that has good flux-to-power ratio and is just critical, but if you build in 25% excess reactivity the situation may change considerably. It is interesting to look at systems that do not need a large amount of excess reactivity.

SPINRAD: I would like to mention a possibility which I alluded to but left out. A type of system which has some attractiveness in this regard is the annular heavy water flux traps. One finds that there is a rather small penalty, perhaps a factor of two in total flux, achieved by making them quite oversize, i.e., making the inner radius quite large and using only a few cm annulus; the core volume still becomes a couple of hundred liters fairly readily. Under these circumstances one has available an enormous flexibility of reactor design. One can use a pressure tube, heterogeneous reactor concept; one can use homogeneous pressure tubes; one can use heterogeneous pressure tubes in which the power density is low enough so that one can contemplate on-line refueling with a cartridge type fuel element. I simply wanted to point out that this is one of the reasons I consider this system rather remarkable.

KEEPIN: With reference to Spinrad's suggestion of the possibility of using a Rover reactor as a source of high flux neutrons, I would like to mention that this is being studied at Los Alamos. Variants of these reactors operated at Jackass Flats, Nevada, should be capable of producing fluxes in excess of 10^{17} for durations of the order of an hour.

deBOISBLANC: I would like to make a few comments concerning Spinrad's remarks about the use of Pu. I do not think we should put this into the class of a dream. As you probably all know, in 1958 the MTR was actually operated on a complete loading of Pu and, as the eutectic 14 weight percent aluminum alloy, it was quite a satisfactory material from the point of view of fuel fabrication technology, cost of fabrication, etc. The operational problems associated with the low delayed neutron fraction will cause no real problem at all and, furthermore, the increment of flux of the order of 18 to 20% implicit in this factor, $\nu(\eta-1)/\eta$, is in fact realized and was experimentally confirmed. There is one factor, however, that does weigh against you in this type of utilization and that is the fast ingrowth of Pu^{240} . Roughly, one can say that the decline of reactivity per MW day is about 2-1/2 times that which is observed in a U^{235} loading for the same reactor. But one can realistically talk about using the world's best burnable poison Pu^{240} by going up to something in the neighborhood of 20 to 26 weight percent Pu. The fuel fabrication techniques which were used in the MTR loading would not be feasible but there are now other methods of doing this using powder metallurgy that were not a proven technology at that time. I would think it would be very encouraging to think of taking a core of approximately 15 in. in length and approximately 75 liters and operating it in the neighborhood of 100 MW. This could compare quite favorably with the advanced conditions that have been promulgated for approximately a factor of two over the HFBR. The operational features of such a reactor and the power density requirement would be of the order of 1.5 MW/liter, which seems perfectly attainable within reasonable heat transfer limitations.

I would like to make one remark with regard to power density. In water

systems, either heavy water or light water, one should not overlook the attainable power densities using vortex cooling. Now, vortex cooling has not been incorporated into the HFIR design, where it was first postulated. That was largely because of the change of the original dimensions of HFIR from 10 in. to 18 in. in length at which time the propagation of the vortex on the tube degenerated into essentially turbulent flow. But, going back to the short tubes, one looks at a double slab, for example, and some other geometries in which power densities of 10 MW/liter seem perfectly attainable and perfectly feasible. This is a little less favorable geometry than the ideal geometry that Spinrad talked about, but I think it would be not at all unreasonable to look at a slab, for example, which is roughly 4 in. thick, and roughly 2' x 2' on a side, and that system would be critical and would be able to operate in sustained vortex flow with pressure drops less than 100 lb/in.². With this, one would be able to extract power at something like 10 MW/liter. Leakage into a reflector region of beryllium (which would have to be removed a little bit from this core in order to allow the water to get in, but still providing a favorable geometry for the beam tubes) is possible in this arrangement and still it can accommodate the coolant. I think that we have not by any means exhausted the possibilities, if one is seriously interested in greater than 10¹⁶ in a small number of facilities, perhaps a single facility.

HENNIG: I would like to describe the Fast Flux Test Facility. My purpose is to elicit your attention to this potentially very interesting high intensity neutron source. If you possibly could, give us your thoughts in such a way that you can influence the design of this machine at this time while it is still feasible. The FFTF is intended as the principal U.S. fast flux test facility for the irradiation testing of IMFBR core parts, materials, and components. Its major mission, therefore, is to be a service for the IMFBR program. Beyond this, we are looking at other applications. We have a study underway right now within Battelle and with six other organizations, which is addressing itself to other functional testing requirements. These organizations are going to consider many variants in design and applications. Steam and helium cooling besides sodium cooling in fast breeder reactors is being studied. In areas of research they will look at solid state physics, physical metallurgy, physical chemistry, nuclear physics, plasma physics, and biology. Conversion processes under consideration besides heat to steam are thermoelectric and thermionic. A wide variety of coolants are being considered for space power application. Several different types of space propulsion systems will be studied. Beyond these, isotope production and fusion power devices will be considered by some of these organizations.

I would like to repeat my interest here is to elicit your advice on how we might bend the FFTF, without perverting its mainline purpose, to be useful to the kind of missions that you are discussing here. I will give you a few of the basic characteristics of a generalized FFTF. Figure 1 shows the spectrum in a nominal core

size of about 600 liters (this is roughly 3' x 3'). For several of the kinds of driver fuels that we have considered, you can see that, depending upon your viewpoint or what your specific interest is, these spectra are very similar or unusually different. In any case, I can point out some of the characteristics other people have noted which are interesting to them. Namely, in the region above 1 MeV we see something like 15 to 20% of the total flux. In the region above 0.1 MeV, which seems to be an interesting threshold for many dislocation type studies in physical metallurgy, we see something between 55 to 70% of the flux. Figure 2 addresses itself quantitatively to some of the discussion that has taken place, and I refer to one of Spinrad's equations, which he quoted earlier, replotting basically flux per unit of $2/3$ MW/liter. In this case, then, you read on the ordinate flux in units of 10^{16} n/cm²/sec, for several different kinds of driver fuel and for a range of core volumes. You will see, as generally recognized, that at constant core power density as the core volume goes up the peak center flux goes up. Also the average flux also goes up and in some cases this makes an interesting difference. The selection of $2/3$ MW/liter has already been cited by many people as far below existing technology for water and recognizing that sodium (the nominal coolant for the FFTF) is a far superior coolant, at least by many standards, one could wonder why we are interested in such a low power density rating for the system. It stems from a variety of things which include fuel cycle cost, the allotment of space in the reactor rather than in beams well outside the reactor to experimental space, and it weights very heavily the probability of successfully achieving any given target that we would aim at at this time. Basically, this reactor, as in the case of most fast reactors, will be capable of operating essentially at any power density that technology will permit, just by making the core smaller at a constant power. The last figure shows the design concept to which we have given the most attention. There are several organizations, besides Battelle, now working on other kinds of designs. The core is about 3 feet in dimension and is nominally surrounded by a basically Inconel type of reflector. The basic concept is a skewed arrangement of two nominally vertical tubes. The tubes are roughly 45 feet long and the big containment sphere is on the order of 110 feet in diameter.

CROCKER: At Harwell we have, among other things, been considering a 100-MW fast flux reactor producing fluxes of about 10^{16} . We have been considering at present the use of Pu carbide fuel instead of the cermet that Battelle has been considering. I am rather interested in the spectra because some of our people have wanted a hard spectra while others have been rather more interested in getting a total energy deposition in samples. There is quite a bit of difference between these two cases, though I do agree that somewhere between 10 keV and 100 keV is the right sort of limit for the production of damage in materials. Our main trouble has been not so much on the actual detailed core design as on the fuel element handling. I think this is the

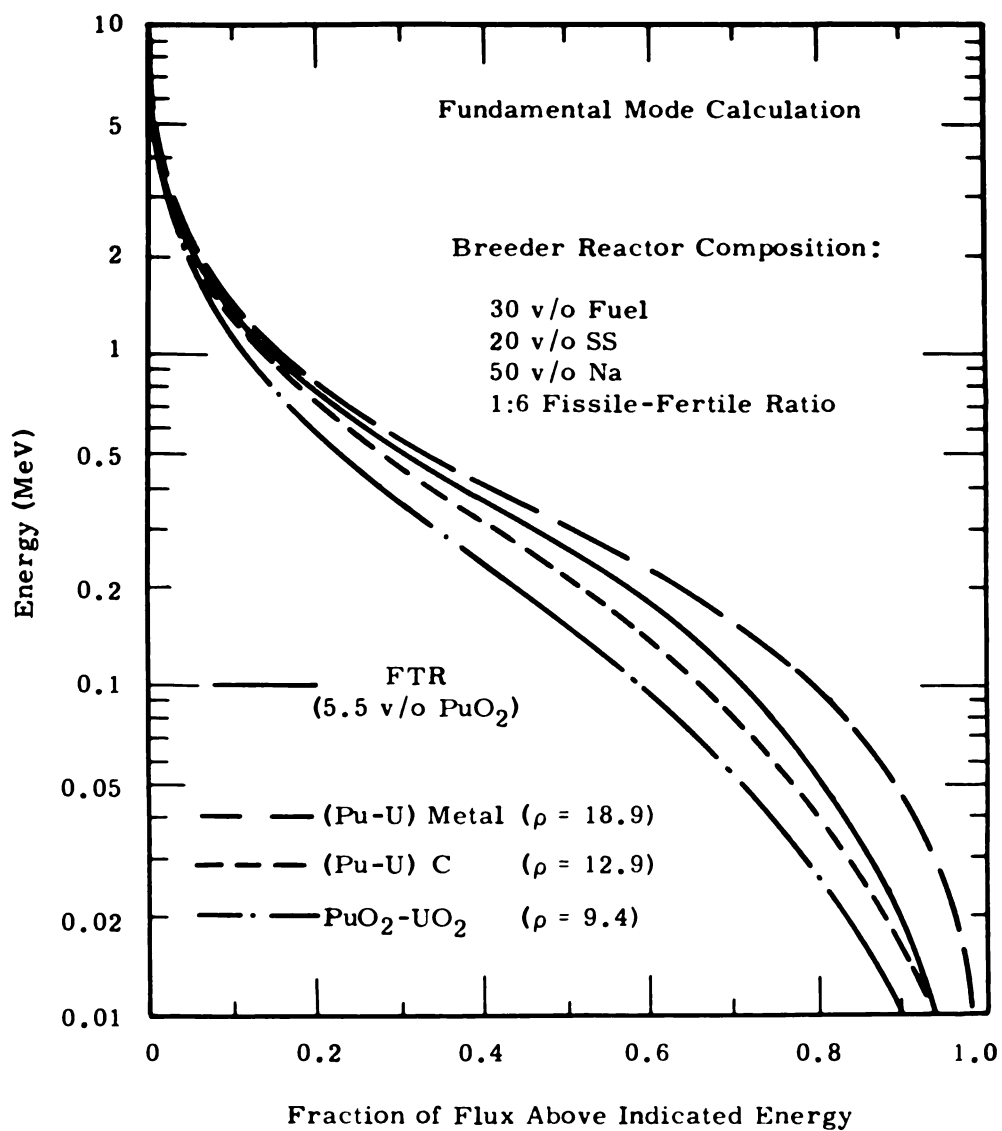
case in this type of reactor and has caused us to look at five or six types of designs already to try to figure which would be the best. We have not made any real decision on this yet.

SPINRAD: I would like to connect Hennig's question with one of the subjects that appeared in my paper very briefly. For several reasons I would be interested in seeing what might happen if an appreciable fraction of the FFTF had its spectrum quite softened down into the truly intermediate reactor range. One reason is technological: that, for accelerated fuel testing even of fast reactor fuels, one can get accelerated burnup, but only a mild distortion of the radial flux pattern is involved. A more interesting reason for this audience is the possibility that we might begin in that way to get some decent experience in the relative cross sections of some of the trans-uranium or transplutonium isotopes which are needed in the heavy isotope production chain. We might very well need a large amount of detailed data on the production process in resonance spectra. I think the FFTF represents the possibility perhaps of getting at some of these data that are just not accessible any other way.

PERRY: I would like to underscore a question that Kouts asked earlier today and which perhaps will not really be answered until the panel discussions on Friday. We have been discussing production of high fluxes in various machines without saying very much about the energy spectrum of the fluxes, and I think perhaps it would be useful for the experimentalists to indicate in more detail than has yet been the case their desire for high fluxes in different energy ranges.

BECKURTS: Well, it is a good question. I would, perhaps a little bit provocatively, say that, as far as the steady-state reactors go, people want just pure thermal fluxes, and nothing else.

In summary, there are indeed interesting possibilities for reaching very much higher fluxes and their technology is not too much beyond our present capabilities. However, these facilities will be expensive, the experiments using them will be very difficult, and certainly a very careful study of the objectives of these facilities is necessary.



**FLUX SPECTRUM IN FTR AND
VARIOUS BREEDER REACTORS**

FIG. 1

CENTER POINT FLUX FOR VARIOUS FUEL COMPOSITIONS

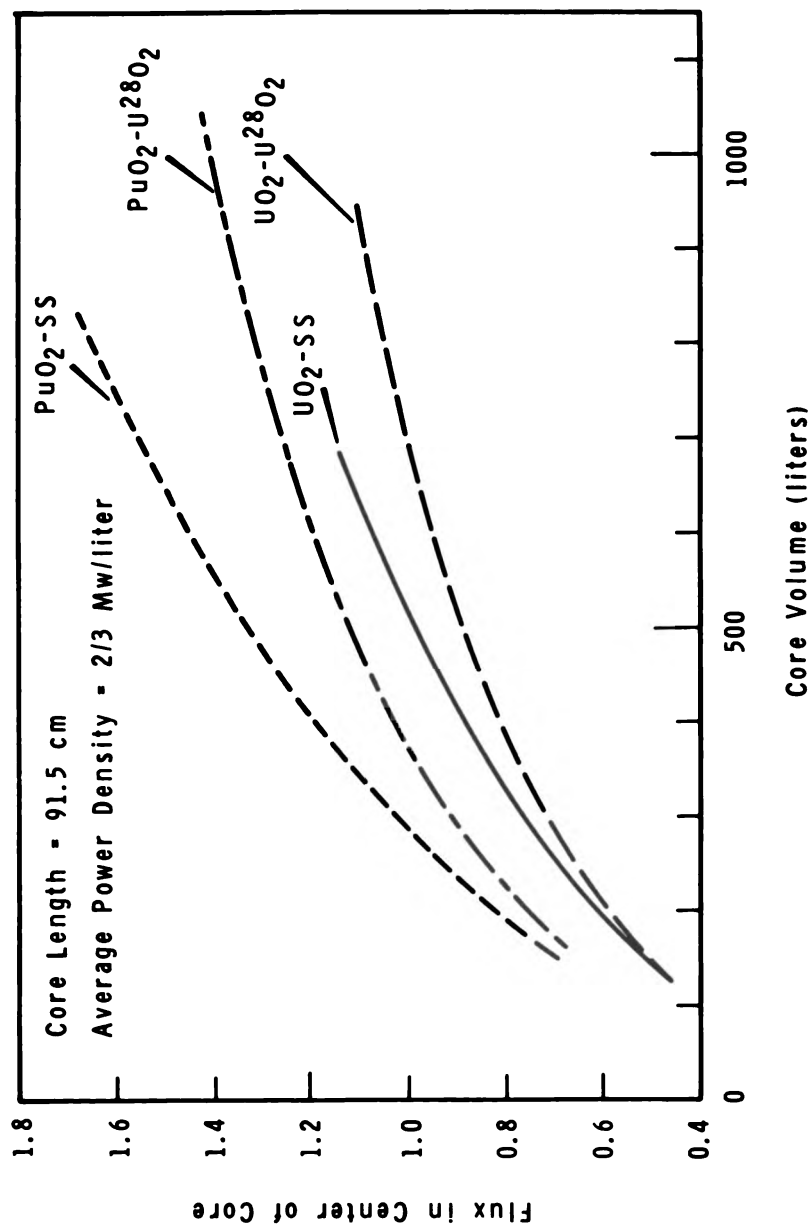


FIG. 2

FAST FLUX TEST FACILITY

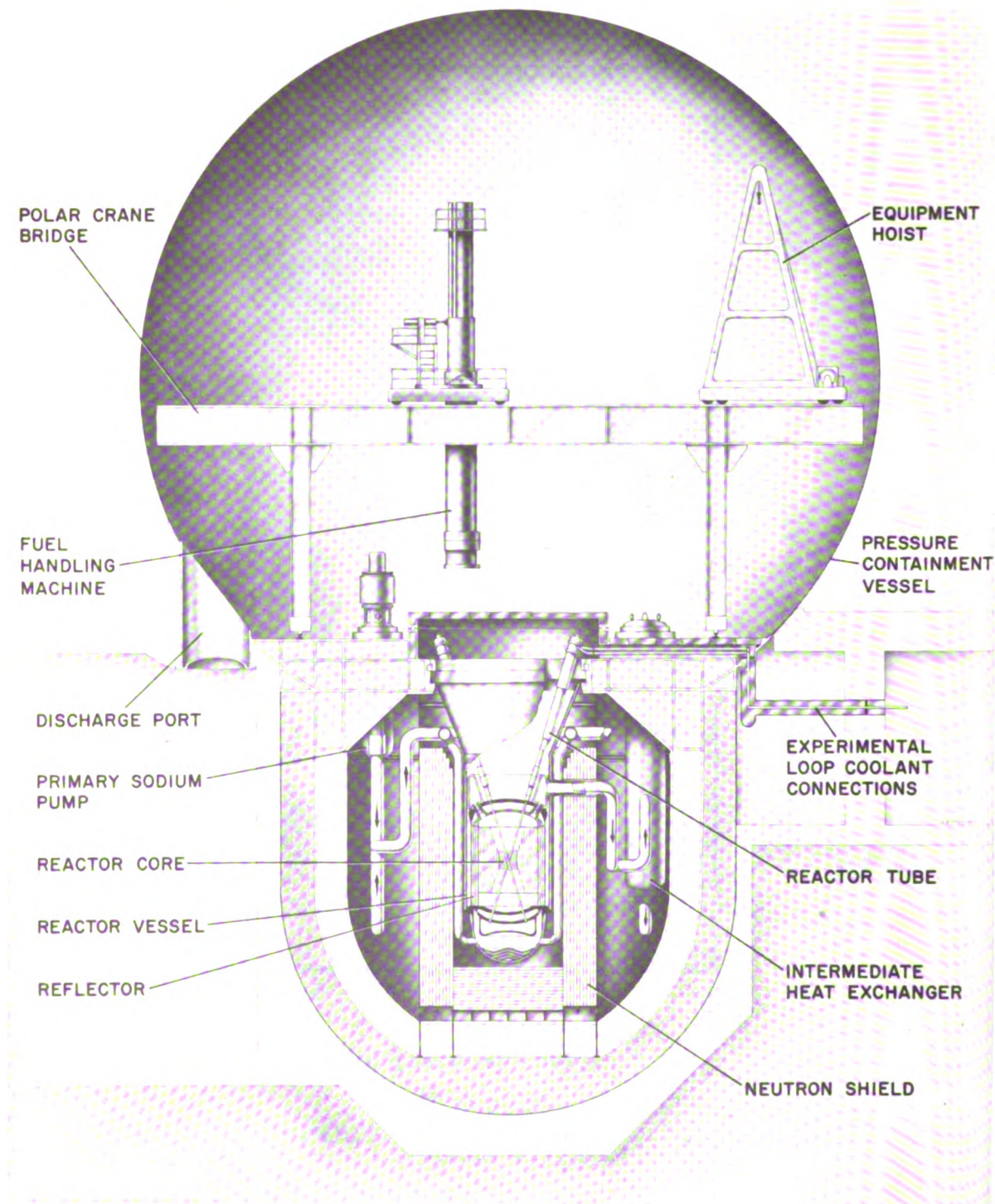


FIG. 3

SESSION III

PULSED FISSION NEUTRON SOURCES

The SORA Reactor : Design Status Report

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Abstract

The design studies for the SORA reactor, which have been underway at the Ispra Center of EURATOM since 1962, are described with special emphasis on detailed studies of general interest for pulsed fission neutron source design. SORA is a periodically pulsed fast reactor with mean power of 0.6 - 1 MW. Reactivity pulsing is done with a reflector block rotating at 50 cycles per second past one face of a compact core. The design has been optimized as a source for thermal and cold neutron experiments using time-of-flight techniques. The overall reactor design is briefly described. Special problems in physics design, reactor core and pulsation device design, dynamics and control, and reactor hazards are discussed.

Contents

1. Introduction
2. General Description
3. Reactor Physics Design
4. Reactor Design
5. Dynamics and Control
6. Reactor Hazards
7. Conclusion

1. INTRODUCTION

The presentation on the IBR reactor at Vienna in August 1961 (1) immediately stirred interest at the Ispra Center, which had only recently been transformed from an Italian nuclear center to the principal establishment of the Common Research Center of EURATOM. To the experimental neutron physicists at Ispra, the fast pulsed reactor seemed to offer a relatively inexpensive way to enter the high flux field when it became time to upgrade their neutron source from the existing 5 MW CP-5 type reactor. To the reactor physicists, the pulsed reactor offered a novel and extremely interesting subject on which to apply and extend their previous experience.

The result was the starting in the Department of Reactor Physics of the SORA reactor design study in 1962. This study has been carried during the past four years to the point where the feasibility and the suitability of construction are now thoroughly demonstrated.

Among the pulsed fission neutron sources under discussion at this seminar, SORA represents a bridge between the highly successful, low power IBR and the high power machines whose development is currently being considered. It is on purpose that the design conditions of SORA do not push the technological limits of materials or heat transfer, as will be required in the advanced machines. Through the choice of the 0.5 - 1 MW power range, we have been able to avoid major design difficulties which will arise in higher power systems while achieving excellent neutron source conditions. For this reason, the SORA design study does not provide guidelines for the ultimate capabilities of pulsed systems. The SORA design has, however, been carried further than the design of any other advanced pulsed fission source. In this paper the results of detailed studies which are of general interest for pulsed fission neutron source design are emphasized.

The detailed design studies on SORA started in 1963. A design contract between EURATOM and the consortium BELGONUCLEAIRE and SIEMENS-SCHUCKERTWERKE lasting from July 1964 to January 1966, led to a design and specifications for the entire reactor plant.

Besides overall coordination of the design project, Ispra has carried out the reactor physics, dynamics and control, safety,

shielding, and experimental facility studies. A critical experiment program for SORA began at Oak Ridge National Laboratory in September 1965, under a cooperative agreement between EURATOM and the USAEC. At the present time, effort at Ispra is centered on analysis of the critical experiments, development testing of special reactor components, safety studies, and detailed plant and facilities layout. According to a projected construction schedule, SORA could be in operation in 1970 if approval is received this year.

2. GENERAL DESCRIPTION

The SORA reactor has been described at earlier stages in the design (2) and at a recent conference (3). Here, we describe briefly the reactor as it looks after the studies made jointly by EURATOM and BN/SSW, with the aid of Fig. 1-3. Fig. 1 gives a cutaway view of the reactor block and shows the general arrangement of the principal parts of the reactor. Fig. 2 gives a schematic layout of the 16 beam tubes. (The levels of the horizontal beam tubes are given with respect to the core midplane.) Fig. 3 shows a horizontal section through the midplane of the core and reflector.

The reactor consists of a concentrated core with a volume of 6 liters containing 123 fuel elements. The fuel is U-Mo alloy with the uranium enriched to 93 % in U-235. These elements are contained in a double-walled vessel connected to the general cooling circuit. The cooling fluid is eutectic NaK. The core has a hexagonal cross section; it is surrounded on all sides, except on one face, by a reflector constituted of an inner layer of W-Ni alloy and an external part in stainless steel. In the reflector are housed six control and safety rods and two containers which hold a hydrogeneous substance, which serves to moderate and to thermalize neutrons for 12 beam tubes (4 horizontal and 2 slanted for each container).

The free face of the core is passed periodically by a Beryllium reflector carried on a rotating arm of Titanium, equilibrated by two shorter arms located at 120° . The shaft of the rotating arm is supported by ball bearings. The total device is contained under helium atmosphere in a housing and cooled by nitrogen flow driven by a ventilator placed at the upper part of the machine.

In Fig. 1 one also sees the fuel handling canal above the core, the coolant circuit entrance and exit lines, the discharge channel of one moderator, the gas cooling circuits for the reflector and control rods and for the pulsation device, the driving motor of the rotor, and the graphite shield. In Fig. 2, one sees the three shutter holes (SH) and the vertical through-tube (IT). In Fig. 3, one also sees the Beryllium block fixation, the finned pulsation device housing, and channels for cooling the reflector and the graphite shield.

3. REACTOR PHYSICS DESIGN

3.1. Pulse Characteristics

Stationary conditions: The reactivity of a pulsed fast reactor (PFR) is a periodic function of the position of some moving part of the reactor. Each period there is a reactivity pulse which takes the reactivity above prompt critical for a very short time; the neutron source coming from delayed neutron precursor decay is multiplied during this time to give a power pulse.

Due to the prompt multiplication of the delayed neutron source between the pulses, all of the power is not generated during the pulses, as would be desirable; a part, called background power, is generated between the pulses. Independent of any details of the reactivity variation, in steady-state pulsed operation the fraction of the energy generated in the pulses is (4):

$$E_p = \frac{\text{energy produced in a pulse}}{\text{energy produced in a period}} = 1 - \frac{\beta}{\epsilon_0} \quad (1)$$

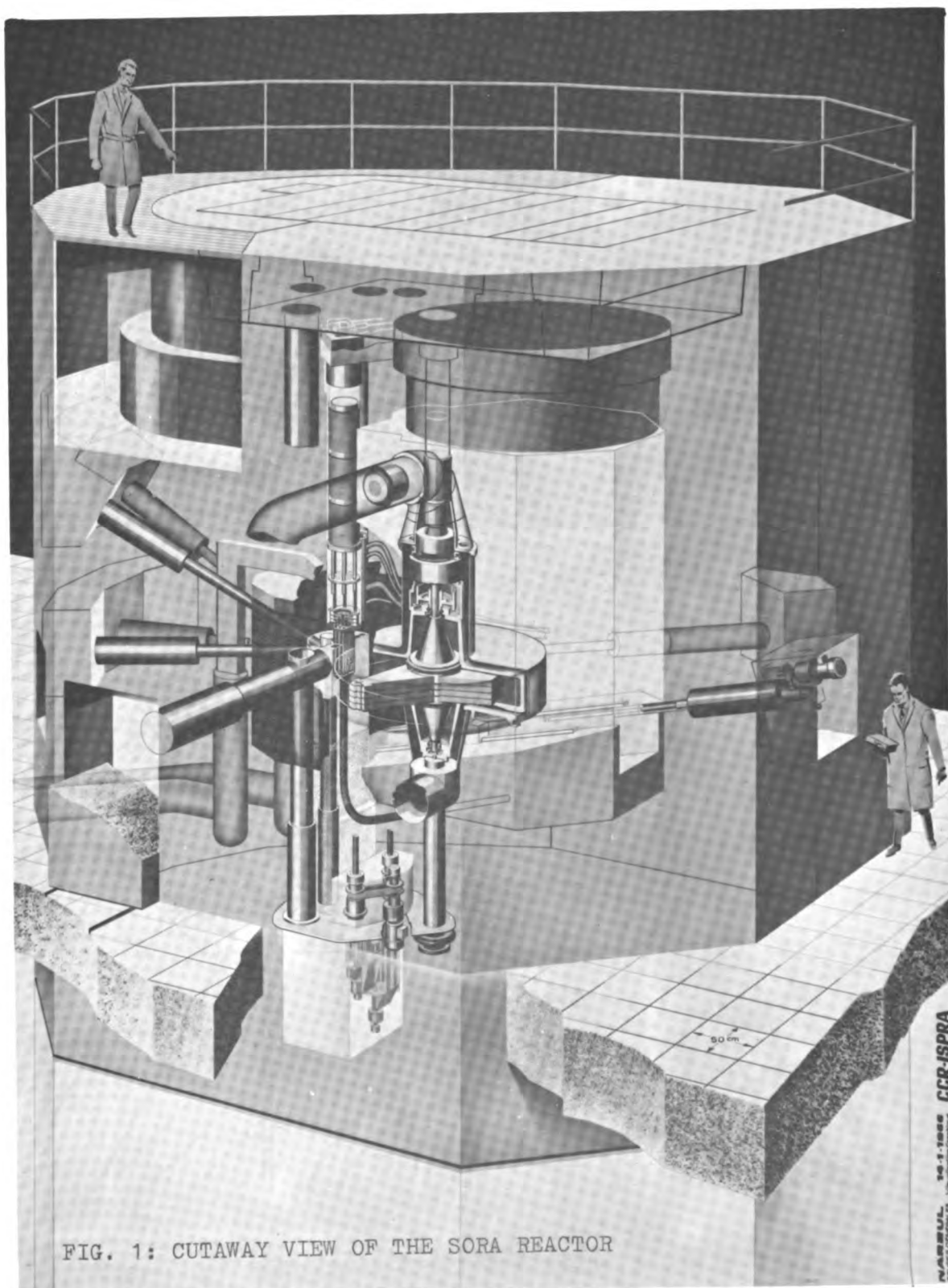
where $\frac{\epsilon_0}{\beta}$ is the absolute value of the average prompt reactivity between pulses, measured in dollars. The energy generated in one pulse, e_p , in steady-state pulsed operation is then:

$$e_p = \frac{E_p \bar{P}}{f}, \quad (2)$$

where \bar{P} is the mean power and f is the pulse repetition rate.

The ratio of mean background power, P_0 , to mean power, as obtained from Eq.(1), is:

$$\frac{P_0}{\bar{P}} = \frac{\beta}{\epsilon_0} \quad (3)$$



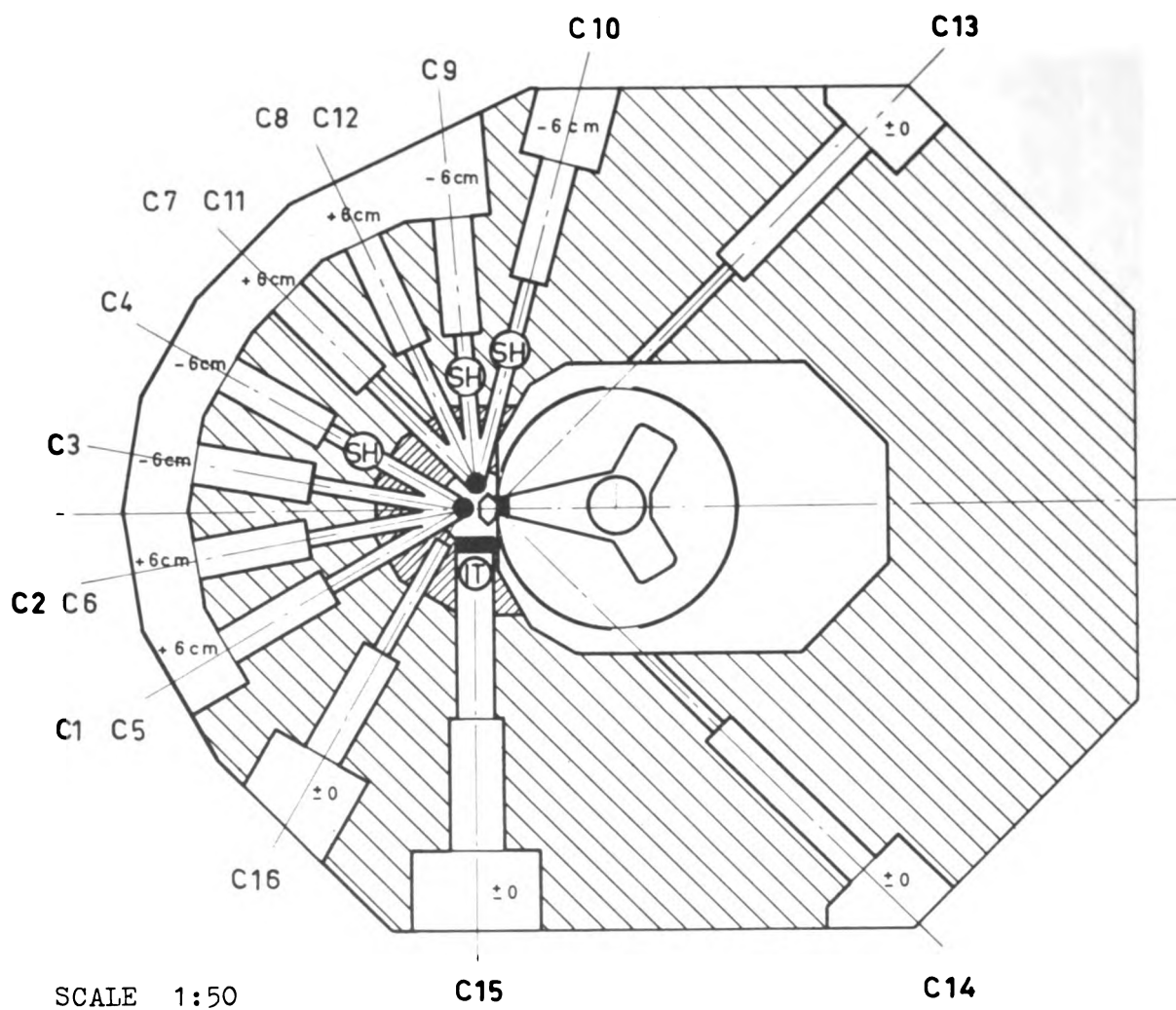


FIG. 2: BEAM TUBE LAYOUT

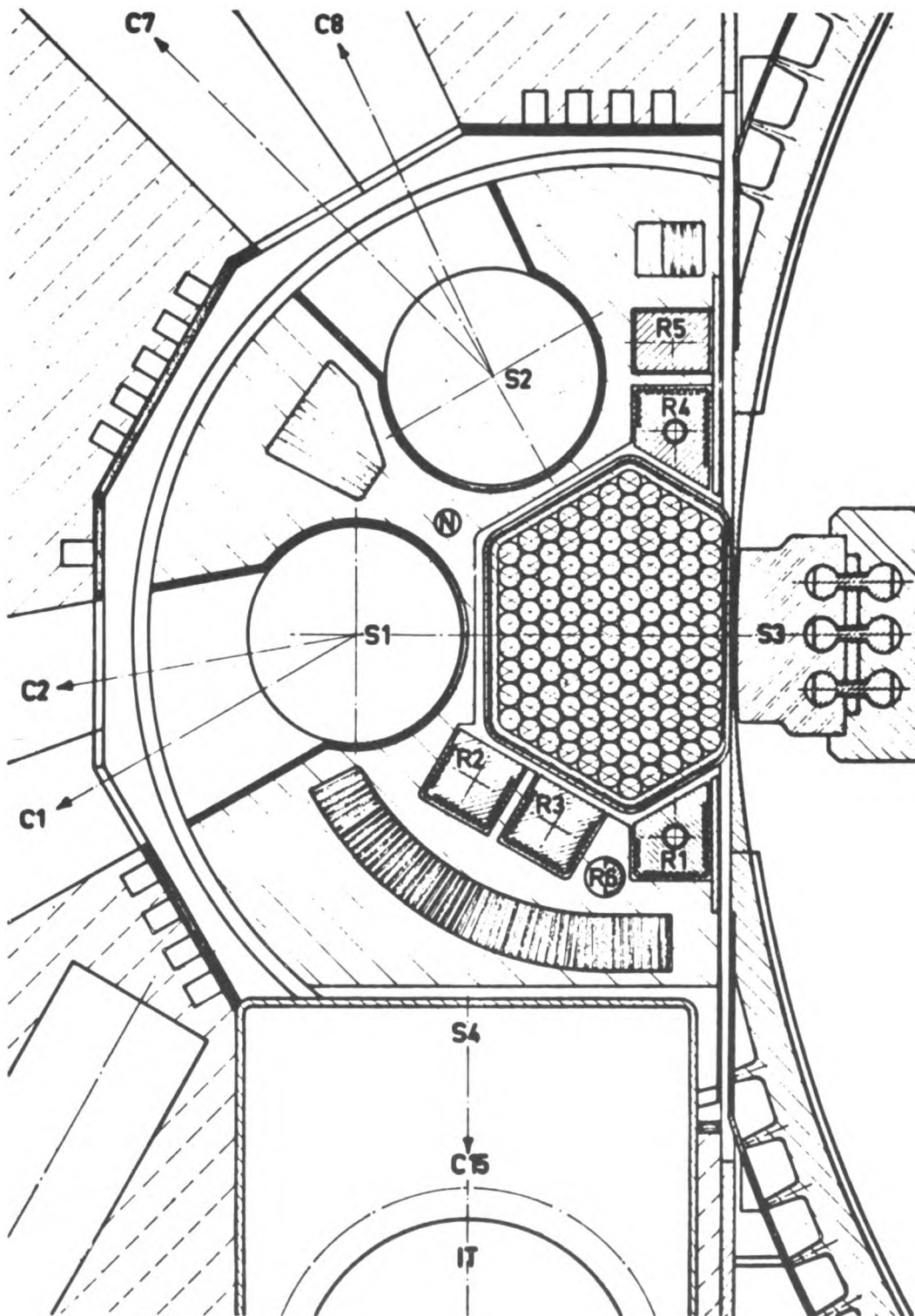


Fig. 3 HORIZONTAL SECTION THROUGH S O R A CORE
AND REFLECTOR

Eqs(1) and (3) show clearly that the performance of a PFR depends strongly on obtaining a large negative prompt reactivity between the pulses in order to concentrate the energy production in the pulses and to reduce the background power level. Since a PFR must be prompt critical during the power pulse, the first requirement on the pulsation device is a large reactivity worth. Since the worth is measured in dollars, this requirement is easier to achieve with Pu-239 or U-233 fuel than with U-235 fuel.

Power pulse characteristics: The energy generated in one pulse is fixed by the so-called pulsed critical condition, Eq.(2). The product of peak pulse power, P_m , and power pulse halfwidth, θ , in a PFR is to a good approximation (4):

$$P_m \cdot \theta = 0.90 e_p. \quad (4)$$

The power pulse in a PFR is nearly Gaussian with a halfwidth given by (5):

$$\theta \cong 2.35 \sqrt{\frac{\tau}{\gamma}} \quad (5)$$

where:

τ is the effective prompt generation time, and

γ is the reactivity insertion rate at prompt criticality.

Thus the second requirement for the pulsation device of a PFR is a high reactivity insertion rate around prompt criticality. The pulsation device is characterized by the function $\epsilon(x)$ giving the reactivity as a function of the position, x , of the moving part. If v is the velocity of the moving part and x_0 the position for prompt criticality,

$$\gamma = v \left. \frac{d\epsilon}{dx} \right|_{x_0} \quad (6)$$

To achieve a large γ requires a high tip velocity, v , and a sharp spatial reactivity pulse, $\epsilon(x)$. The peak of the reactivity pulse can be adequately approximated by a parabola (4,5):

$$\epsilon(x) = \epsilon_m - \alpha x^2, \quad (7)$$

so that the reactivity insertion rate can be characterized by the parameter α .

In steady-state pulsed operation of a PFR, the power pulse

width can be rewritten as (4,5):

$$\theta \cong c \left(\frac{\tau}{\alpha v^2} \right)^{\frac{1}{3}}, \quad (8)$$

where c is approximately 1.5 and is weakly dependent on the power pulse size. This equation summarizes the fundamental limitation on power pulse width in a PFR. The value of $(\tau / \alpha v^2)^{1/3}$ achieved in the IBR is about 28 μ sec; in SORA, it is about 35 μ sec. The increase in SORA results from almost a factor of two in α due to the use of an external moving reflector rather than a central fuel piece, and about a 50 % increase in τ due to the location of the moderators very close to the core and to the NaK coolant and the Be reflector. The velocity has been increased for SORA by 23 %.

Table 1 gives the estimated values of reactor parameters and the calculated pulse characteristics for the SORA reactor.

Table 1
SORA Estimated Power Pulse Characteristics

Fuel	U-235	Pu-239
τ , nsec	20	24
$\frac{\tau}{P}$	6	19
α , m^{-2}		6
f , sec^{-1}		50
v , $m \ sec^{-1}$		283
\bar{P} , kW	600	1000
P_m , MW	170	340
$\frac{P_m}{P_o}$	1700	6500
θ , μ sec	52	51

The left column represents the initial operating conditions with U-235 fuel and 600 kW power; the right column represents the ultimate operating conditions with plutonium fuel and 1 MW power. The principal advantage of changing to plutonium fuel will be in the improved peak to background ratio.

Operation as accelerator pulsed fast reactor (APFR)

Consideration has also been given to the operation of SORA as an accelerator pulsed fast reactor (APFR) to reduce the power

pulse width. This is considered at the present time as a possible mode of operation after sufficient experience is gained with SORA as a PFR, and provisions are being made in the design of the buildings and installations for this eventuality. Although our studies to date on the accelerator, beam handling and target have been of a very preliminary nature, they have made us aware that the conversion of SORA to APFR operation will be sufficiently expensive and complicated that an extensive experimental program based on the improved pulse characteristics will be required to justify the conversion.

The steady-state pulsed condition, Eqs(1),(2), and (3), apply also for APFR operation. In an APFR, the product of peak power and halfwidth is approximately:

$$P_m \cdot \theta \cong 0.70 e_p, \quad (9)$$

if the LINAC pulse length is short compared to the power pulse width. This condition is of interest for several reasons; in particular it permits the amplification to be maximized, thus reducing the required accelerator mean beam power or target power.

The power pulse length for an APFR in the short electron pulse length limit is (4):

$$\theta = 0.7 \cdot K \frac{e_p \tau}{P_b t_a} \quad (10)$$

where

P_b is the accelerator beam power,

t_a is the accelerator pulse length, and

K is a relative efficiency of neutron production from fission and from electrons, defined as:

$$K = \frac{\nu C_f}{\omega C_e}, \text{ with}$$

νC_f = fundamental mode fission neutrons produced per unit reactor power, and

ωC_e = equivalent fundamental mode fission neutrons produced per unit electron beam power.

The mean accelerator beam power or target power, P_t , is:

$$P_t = P_b t_a f, \quad (11)$$

and the APFR fundamental mode amplification, A , is in the short electron pulse length limit:

$$A = \frac{\theta}{0.72} \quad (12)$$

Table 2 gives estimated pulse characteristics for the 1 MW SORA in APFR operation. The first column shows the characteristics for an accelerator beam power - pulse length product of 1000 MW- μ sec. Such an accelerator is about the largest reasonable to consider for coupling with SORA. The reduction of θ in going from PFR to APFR operation would be from 50 to 6 μ sec. The pulse width could be reduced below 6 μ sec with reduced mean power; for example, for $\theta = 2$ μ sec, the mean power would be reduced to about 300 kW. (The peak power would remain about the same.)

Table 2

SORA APFR Estimated Power Pulse Characteristics

$P_b \times t_a$, MW - μ sec	1000	300	
f , sec ⁻¹	50	50	150
P_t , kW	50	15	45
\bar{P} , MW	1	1	0.73
P_m , MW	2100	640	640
$\frac{P_m}{P_o}$	36,000	6500	8500
θ , μ sec	6.3	20	5
A	360	1150	290

The second and third column show the characteristics for a two mode operation of the APFR: 1 MW mean power with 50 pulses/sec and 20 μ sec pulsewidth; 5 μ sec pulsewidth with 150 pulses/sec and lower mean power. The first operational mode would be suitable for cold and thermal neutron experiments for which low pulse frequency is desired. The second mode would be suited for higher energy neutron experiments for which the higher pulse frequency is acceptable. The reduction in required accelerator beam power brings the accelerator characteristics closer to present technical capabilities. However, this gain is made at the expense of reducing the overall efficiency with which the neutron source is utilized.

As would be expected, it is difficult to optimize APFR operation conditions as regards pulse width and pulse frequency for all types of experiments at the same time. This last consideration as well as the expected operational and maintenance difficulties are the reasons for our wait-and-see attitude about future conversion of SORA to APFR operation.

3.2. Physics Design of Reactor

The physics design of SORA is based on an extensive calculational program at Ispra and on a critical experiment program performed jointly by EURATOM and the USAEC at Oak Ridge National Laboratory.

Because the principal physics design problems are related to the very heterogeneous system of reflectors and moderators surrounding the highly enriched small core, precise methods of calculation have been used. Time independent stationary and quasi-stationary neutron transport problems (for which it is assumed that an equilibrium neutron distribution exists when the reactivity changes rapidly) have been solved by the S_N method (using the DTK, 2DXY, and TDC codes (6)), and by a new Monte Carlo code - called TIMOC - which was developed at Ispra (7). Special geometry routines written for TIMOC allow a very detailed mockup of the reactor, especially the moving reflector, the vessel in the window region, the control and safety rods and the moderators.

Even with the use of these advanced methods of calculation, it was decided that some experimental checks were necessary. We were happy to find that the USAEC and ORNL were interested and able to perform these measurements, and in May 1965 the USAEC and EURATOM signed a Memorandum of Understanding Regarding Cooperation Pertaining to the SORA Reactor. The first critical measurement was performed on September 28, 1965, and most of the initially planned experiments were completed by early-1966. During the Spring and Summer of 1966, further refined measurements have been planned based on the very successful and interesting first results.

The critical experiments are being performed on a rather exact geometrical mockup of the SORA design. Reflector materials were chosen to cover those planned in the design and also to provide further calculational checks. The core, which is less

a serious calculational problem, is mocked up by highly enriched U-235 rods located in an iron matrix (which simulates the Mo, NaK and cladding).

Calculations of reactivity for measured critical configurations have shown good agreement. For a so-called "Nominal Design", which has Be moving reflector and safety rods and paraffin moderators, calculated values of k_{eff} were:

$$\begin{aligned} &1.006 \pm 0.004 \text{ from TIMOC, and} \\ &0.994 \pm 0.002 \text{ from 2DXY.} \end{aligned}$$

In the TIMOC calculations, the 26 group Russian cross section set (8) has been used. Some not unexpected problems have arisen with the proper iron cross sections to use in the reflector regions. For the quoted value, the upper 15 groups of the Russian iron cross section set were replaced by the 20 group Yiftah and Sieger cross sections (9), and non-self shielded iron cross sections were used within the first 2.5 cm of the core. (The use of self-shielded iron cross sections in this region reduced k_{eff} by 1.6 %; the Russian self shielded cross sections increased k_{eff} by 1.4 %.)

For the S_N calculations, the cross sections were based on the 18 group LASL set (10), reduced to 6 groups; the S_4 approximation was used. The 2DXY value is the average of values calculated for two halves of the system. (The value on the side with the sidemoderator was 0.985, and on the side with control rods, 1.003.) For these calculations, two sets of region and energy dependent axial bucklings, representing upper and lower limits, were estimated from TDC calculations. Comparison with three measured configurations indicated that the larger set of bucklings gave the best agreement.

Moving reflector worth and reactivity pulse shape:

The principal emphasis of the experimental program has been placed on measurements of reflector block worth and reactivity pulse shape for various materials and configurations (11). Table 3 gives experimentally determined worths and α 's (12) and TIMOC worth calculations. In these measurements, the Nominal Design was used (see above) and the mock-up window between the core and the moving reflector was made up of 5 mm Fe/ 2 mm air/ 3 mm Fe/ 3 mm Al/ 4 mm air. The α values are obtained from a parabolic fit to the upper 20-30 % of the reactivity pulse shape.

Table 3

Measured and Calculated Moving Reflector Parameters

Material	Beryllium			
Width, cm	11	16	21	26
Worth, \$				
Measured	6.0 \pm 0.2	8.8 \pm 0.2	10.9 \pm 0.3	12.1 \pm 0.2
Calculated	6.2 \pm 0.3	9.0 \pm 0.3	11.0 \pm 0.3	-
α, m^{-2}	5.5 \pm 0.6	6.1 \pm 0.6	6.2 \pm 0.6	4.7 \pm 0.5

Considering first the reflector block worths, the agreement between calculated and measured values for the beryllium blocks is excellent. A further series of measurements was performed with iron reflector blocks. Calculated worths for these cases were all higher than measured worths, which must be attributed to difficulties mentioned above with adequately representing the iron cross sections.

Up to the present time, attempts to calculate alpha by various methods have not been successful. Therefore, the measured α values were new information. The increase in the sharpness of the pulses for large Be blocks was not expected and clearly suggests increasing the Be block width above the 11 cm previously foreseen. Additional measurements are currently being performed to get a better physical understanding of the parameters influencing the reactivity pulse shape of a moving reflector in these configurations. On the basis of these measurements, the moving reflector block and window designs will be modified in a new compromise between improved pulse characteristics (principally, lower background) and mechanical design.

The experimental program also included many measurements on control rod and scatterer worths and on various reactivity effect measurements. Analysis of these results is continuing, and to date has generally shown reasonable agreement. Publication of both experimental and calculations results will be made soon.

Prompt generation time: The formulas in section 3.1 for the power pulse characteristics are based on a one-energy group point neutron kinetics model. The principal kinetics parameter of this model is the prompt generation time, τ . In the SORA reactor, the

definition of τ is complicated by the presence of neutrons with lifetimes on the order of and longer than the power pulse width. These are primarily neutrons which leave the core and are slowed down in the homogeneous moderators. A layer of boron is placed between the moderators and the core which is very effective in preventing the return of very low energy neutrons to the core and partially effective in preventing the return of epithermal energy neutrons.

A 3-energy-group, 2-space point kinetics model has been developed based on the 2DXY code, to evaluate kinetics parameters, power pulse characteristics, and low energy neutron moderator pulse characteristics (13). For determining the power pulse characteristics, only the first two energy groups (10 MeV - 454 eV and 454 eV - 0.414 eV) need be considered. The kinetics equations are:

$$\frac{dn_1}{dt} = \frac{k_1(1-\beta)-1}{l_1} n_1 + \frac{k_2(1-\beta)}{l_2} n_2 + S \quad (13)$$

$$\frac{dn_2}{dt} = \frac{p_{12}n_1}{l_1} - \frac{n_2}{l_2} \quad (14)$$

The physical meaning of the parameters is clear; the evaluation has been done using 2DXY flux and adjoint calculations. Typical values for the SORA design are:

$$\begin{aligned} l_1 &= 19.3 \text{ nsec}; & p_{12} &= 0.0011 \\ l_2 &= 4.0 \mu\text{sec}; & k_2 &= 1.40. \end{aligned}$$

In simple terms, about 0.1 % of the neutrons causing fission have a generation time of about 4 μ sec. The feedback of these fission neutrons to the first group broadens the power pulse, or in other terms causes an increase in effective prompt generation time. For example, interpretation of the pulse width for the above 2 group case using the one group model yields an effective one-group τ of 22 nsec, compared to the 19.3 nsec for l_1 . This result indicates that the boron shield is sufficiently effective in preventing broadening of the power pulse by epithermal neutrons.

It is planned to operate the mockup in periodically pulsed condition. The data obtained will provide a very useful check

on the adequateness of the method outlined above for predicting the power pulse characteristics.

3.3. Physics Design of Neutron Sources

The PFR poses specific problems for the design of the neutron sources which are somewhat similar to the problems encountered at other pulsed neutron sources. SORA, in contrast to usual LINAC operation, has been optimized for cold and thermal neutron experiments (14, 15, 16).

For the layout of the neutron sources a practical compromise has to be found between the number of neutrons per pulse and the pulse width. To maximize neutron intensity, the sources are placed as close as possible to the core. The pulse width depends primarily on the neutron relaxation time in the source since the reactor pulse width is fixed.

For cold and thermal neutrons the reactor pulse width of 50 μ sec does not constitute a severe problem because practical pulse widths are always of the same order or larger. Thermalization time and epithermal leakage considerations suggest that relaxation times between 40 μ sec and 120 μ sec be used. Furthermore, pulse widths which can be obtained with these relaxation times allow, together with the chosen pulse repetition rate, sufficient energy resolution to be obtained in many experiments, to that a reduction of pulse width by choppers is not required.

Relaxation times of this order are obtained in the SORA moderator geometry by moderator blocks of 0.7 to 2 liters of water, hydrogen or similar substances. The maximum diameter of the moderator is 14 cm (including the thermal insulation).

The problem of efficiently extracting the neutrons from the moderator is more difficult. Therefore, the horizontal beamholes were arranged in 2 layers in a way that effectively 4 moderator surfaces can be optimized for 2 beam tubes each.

For cold neutrons a heterogeneous design was chosen. One half cylinder consists of homogeneous material and the other half (towards the beamtubes) is made of Beryllium. In this way a relatively high thermal flux is maintained at the homogeneous moderator surface, so that the cold leakage flux is strongly enhanced. This type of source can supply cold neutrons to different beam tubes at rather large angles. Moderator temperatures between 10 to 70 $^{\circ}$ K are foreseen for this source,

depending on the moderator chosen.

For thermal neutrons (with energies larger than 5 meV), a reentrant hole geometry is advisable. However, for beam cross sections comparable to the moderator size, good conditions are difficult to achieve, particularly for several beam tubes in different directions. As moderator material for thermal and epithermal neutrons, water or Zirconium Hydride will be used at temperatures between -5°C and 400°C .

For eV and keV neutrons the possible sources are the movable Be block, the stationary reflector and special sources in the 30 cm diameter beam tube. The reactor pulse defines the source pulse width for these neutrons because the slowing down times and even more the relaxation times of these neutrons are short compared to the reactor pulsewidth.

Experimental program: In order to verify the cold and thermal source design concepts and to arrive at optimized source geometry and materials an experimental program was started at the Ispra 1 MeV Van de Graaff. So far results are available on 5 source types at room temperature and at liquid nitrogen temperature. Three source containers were placed in a cryostat which was housed in a block of steel to simulate the SORA geometry. Each source container has an outer diameter of 11.6 cm and an overall height of 15.3 cm. Container A is a simple cylinder; container B contains a square reentrant hole of cross section $5 \times 5 \text{ cm}^2$ and depth 5.4 cm; container C is divided into two independent half cylinders. Geometry and material compositions studied are:

- A : container A filled with H_2O
- B1 : container B filled with H_2O , reentrant hole empty
- B2 : container B filled with H_2O , reentrant hole plugged with $5 \times 5 \times 5 \text{ cm}^3$ Be block
- C1 : container C with front half filled with H_2O , back half empty
- C2 : container C with front half filled with H_2O , back half filled with Be.

Details of the experiments and results are given in Ref.(17). The following summarizes some of the main results:

- The cold neutron leakage fluxes (energies of 5 meV and be-

low) at 77° moderator temperature are highest (and about the same) for B2 and C2. The peak fluxes are about 2.5 times higher than for C1 and for A.

- For all cold sources without Be the emerging flux peaks at 10 to 11 meV with the largest flux at this energy for B1, which is 2 times larger than A and 3 times larger than C1.
- For all room temperature sources (without Beryllium) the flux peaks between 28 and 30 meV. B1 has the largest flux, which is 2.4 times larger than C1 and 2.8 times larger than A.
- The ratio of the peak fluxes of A to C1 is for room temperature 1 : 1.17 and at 77 °K 1 : 0.65. The asymptotic relaxation times in μ sec were for A and C1, respectively: at room temperature, 98 and 58; and at 77 °K, 148 and 95. This indicates that due to the slow thermalization process in cold ice too many neutrons are lost before thermalization is completed if the relaxation time is only 95 μ sec.
- The relaxation time for C2 at 77 °K was 115 μ sec.
- The relaxation time for B1 at 300 °K was 66 μ sec.

These results support the design philosophy chosen; however, more experiments are necessary on both the cold and thermal sources mainly concerning moderating materials in the first case and geometries in the second case. Furthermore, the effects of the cooling provisions in the moderator will be studied. The cold moderators will be cooled with nitrogen or helium as long as a solid moderator is required for reactor safety reasons. The cooling load is about 400 Watts for geometry C2. This cooling load is smaller in a PFR than in a high flux reactor, but the gain is not in proportion to the mean powers partly because a larger volume cold source is required in the PFR. It is planned to study a heterogeneous cold source arrangement with a small fraction of cold moderator for SORA in the future.

A further series of measurements with liquid and solid hydrogen sources is currently being prepared at Ispra. The first aim of these experiments is the measurement of relaxation times and neutron yields for different energies and ortho- and para-hydrogen concentrations.

Theoretical studies: The study of the production of cold, thermal and epithermal neutrons in the complicated SORA geometry is extremely difficult theoretically and is better suited for experimental study. The details of the moderator designs do not affect the reactor operation and improvement of the moderators will be made on the basis of operational experience. For these reasons, relatively little calculational effort has been given to the moderator design to date, and emphasis has been placed more on understanding and limiting the interaction of the moderators with the core. In the critical experiment program, measurements of reactivity effects of various simulated moderator geometries have been made.

Calculations of thermal neutron fluxes and pulse characteristics have been limited to one thermal group. Calculations performed include:

- steady state maximum and leakage thermal group fluxes obtained from 6-group 2DXY calculations;
- average thermal group pulse characteristics based on kinetics parameters determined from 2DXY calculations;
- Studies of time dependent slowing down in water slabs of various thicknesses by the j_N method.

The integrated thermal neutron flux over a second leaking into a beam tube has been obtained directly from the 2DXY steady state calculations. Using S_0 (and 6 energy groups), the angular flux leaving the moderator in the direction of the beam tubes per unit core power has been estimated from specially obtained intermediate results in the code. Multiplying this by 4π gives a fictitious isotropic source flux, ϕ_s , feeding the beamtubes per unit core power. This is equivalent to the value of integrated leakage flux (neutrons/cm²) per unit energy production. Multiplying by the energy in a pulse, e_p , gives the integrated leakage flux per pulse (n/cm² per pulse), and multiplying by the pulse frequency gives the integrated leakage flux per second (n/cm² per second).

The product of the peak leakage flux, ϕ_{sm} , and the half-width of a thermal neutron pulse, θ_s , are related to the integrated leakage flux per pulse by:

$$\phi_{sm} \cdot \theta_s = C \cdot \phi_s \cdot e_p \quad (15)$$

where C depends on the ratio of thermal group lifetime to power pulse halfwidth and is approximately 0.8 when these are about equal.

From an S_8 2DXY calculation on the SORA geometry for a central moderator at room temperature with half cylinder H_2O , the value of ϕ_s was estimated to be:

$$\phi_s = \begin{cases} 7.7 \times 10^{12} \text{ n/cm}^2/\text{sec/MW} & : \text{U-235 fuel} \\ 9.3 \times 10^{12} \text{ n/cm}^2/\text{sec/MW} & : \text{Pu-239 fuel} \end{cases}$$

From Eq.(15) and with the assumption that Θ_s is 75 μsec , we estimate for this thermal neutron source:

$$\phi_{sm} = \begin{cases} 0.8 \times 10^{15} \text{ n/cm}^2/\text{sec} & : 600 \text{ kW} - \text{U fuel} \\ 1.9 \times 10^{15} \text{ n/cm}^2/\text{sec} & : 1 \text{ MW} - \text{Pu fuel} \end{cases}$$

This geometry is not optimum. Results from the experimental program indicate that up to a factor 2 can be gained in the integrated and peak leakage fluxes with a reentrant geometry.

A fictitious epithermal neutron source has also been estimated from the S_N calculations for the half cylinder H_2O central moderator. The peak flux at 1 eV is approximately:

$$\phi_{sm} = \begin{cases} 2.3 \times 10^{14} \text{ n/cm}^2/\text{sec/eV} & : 600 \text{ kW} - \text{U fuel} \\ 5.3 \times 10^{14} \text{ n/cm}^2/\text{sec/eV} & : 1 \text{ MW} - \text{Pu fuel} \end{cases}$$

with a pulse half width of 50 μsec .

The two-point three-group kinetics model (13) mentioned in section 3.2 has been used to calculate thermal group pulse shapes using kinetics parameters determined from 2 DXY flux and adjoint calculations. Typical results are given in Fig. 4 for a design in which the power pulse half width was 52 μsec , the thermal group lifetime was 51 μsec in the central moderator and 38 μsec in the side moderator, and the ratio of mean to background power was 5. In an attempt to reduce the halfwidth, the size of the moderator was decreased in this design. This reduced the peak thermal flux in the side scatterer relative to the central scatterer by about 2.5. (For similar large moderators, the peak thermal flux in the side scatterer is about 15 % lower.) The resulting thermal pulse width reduction was only from 84 μsec for the central scatterer to 74 μsec for the side moderator.

This illustrates the design problem for the moderator in a PFR: to effectively moderate the neutrons from the core while obtaining a short thermal neutron pulse width. To study this problem further, the new j_N method (in the j_5 approximation) has been applied to the time dependent slowing down of neutrons incident on water slabs of varying thicknesses (18, 19). The vector flux as a function of time and space for a 7 energy group (1 thermal group) model has been calculated. The boundary source spectrum was an approximate simulation of the spectrum incident on the SORA moderators, obtained from the S_N calculations described above. To account for the finite height and width of the water slab, a set of energy dependent bucklings was introduced, estimated from spatial distributions obtained from the S_N calculations. The time behaviour of the source used was a Gaussian with 50 μ sec half-width. The calculated peak values and halfwidths of the vector thermal flux pulses ($E \leq 0.414$ eV) directed towards the beamhole at the outer boundary of (90 % full density) water slabs of thicknesses from 3.5 cm to 10 cm are shown in Fig. 5. These results show a rapid increase in the peak value in the range of 3.5 - 7 cm thickness, and a saturation at 10 cm. (Other calculations have shown the decrease of the peak value beyond 15 cm.) The halfwidth increases almost linearly with slab thickness. From these calculations, one could conclude that (90 % density) water slab thicknesses between 5 and 7 cm are most interesting; below 5 cm, the loss in peak intensity is too great to compensate for the small halfwidth reduction, while above 7 cm the gain in peak intensity is too small to compensate for the increase in pulse halfwidth. Of course, these results apply to an idealized system, and the conclusions depend on the power pulse width of 50 μ sec. In APFR operation there would be more incentive to sacrifice peak intensity for shorter halfwidth.

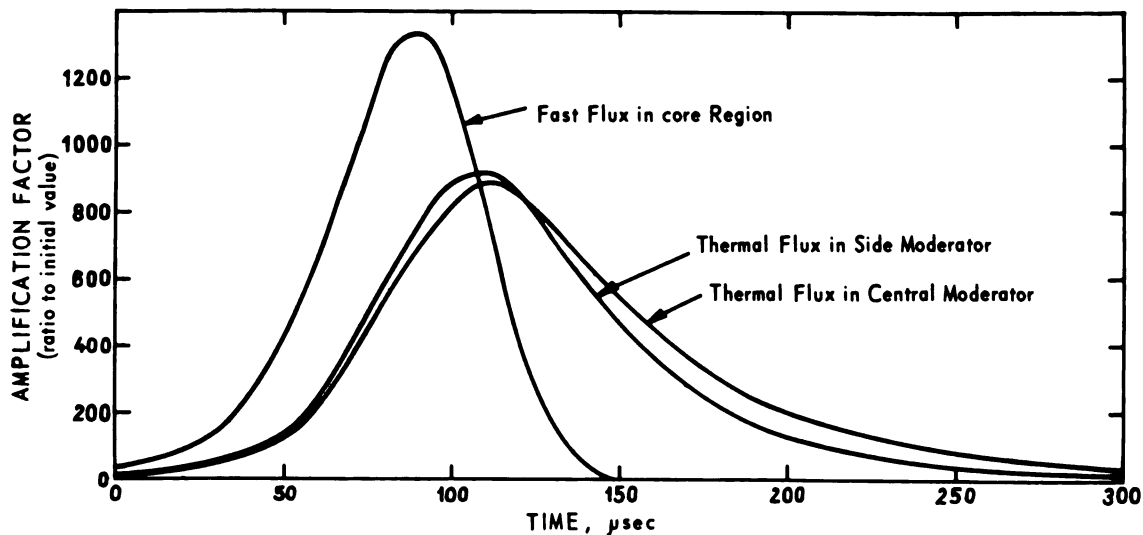


FIG. 4: FAST AND THERMAL NEUTRON FLUX PULSES IN SORA

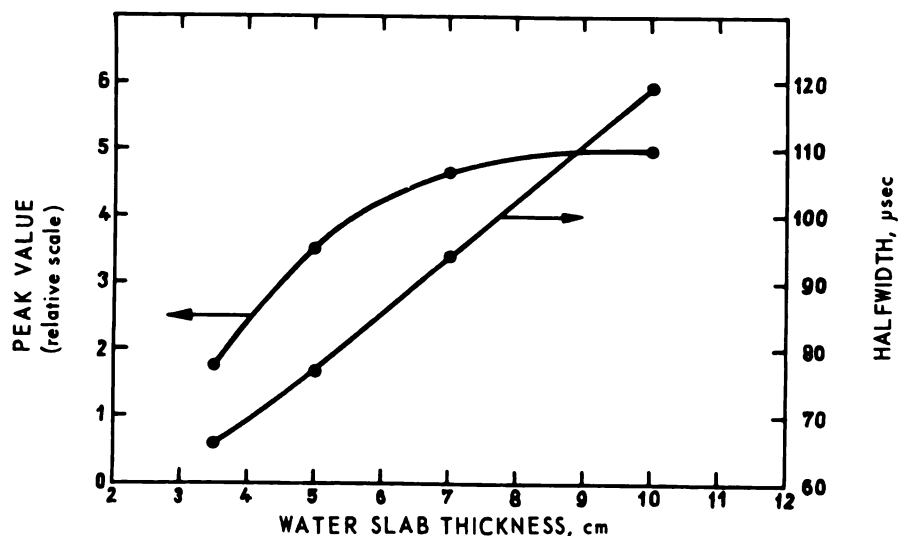


FIG. 5: PEAK VALUES AND HALFWIDTHS OF THERMAL LEAKAGE FLUX PULSES AS A FUNCTION OF WATER SLAB THICKNESS

4. REACTOR DESIGN

4.1. Reactor Materials

Fuel: Low delayed neutron fraction and high fissile atom density requirements in a pulsed reactor suggest the use of high percentage Pu alloy (or U-233) fuel. However, considerations of availability, experience and safety led EURATOM in 1964 to choose highly enriched Uranium-Molybdenum alloy fuel for the initial core loading of SORA.

A further requirement on the fuel material is acceptable behaviour to a specified burn-up. Under expected SORA operating conditions, the average burn-up will be 0.2 % heavy atoms (at 600 kW power) so that fuel lifetime is not a major problem, but there will of course be economic advantages to having a high fuel burn-up. Initially it was thought that a reliable, fast negative reactivity coefficient due to axial thermal expansion of the fuel was an important safety consideration, but the studies described in section 5.3 have shown this is not true. Fuel movement due to irradiation damage or to pulsing could be a problem but its importance is difficult to assess prior to operational experience; it would be a safety problem only if it occurs on the timescale of the pulsing, e.g. within 20-40 msec.

U-Mo alloy in the range 9-10 w/o Mo was chosen for SORA because of the considerable experience available and because information on the behaviour in a fast reactor spectrum would be obtained in the Dounreay and Fermi fast reactors. Irradiation of this alloy in the Dounreay reactor has shown considerable gross cracking of the fuel at burn-ups from 0.25-0.30 atom percent (20). The cracking and localized fuel melting occurring in the Dounreay driver fuel may be found acceptable in a pulsed fast reactor, but they are to be avoided if possible. While waiting for irradiation results on the U-10 w/o Mo fuel in Fermi (where cracking is not expected), we are looking with interest at the results with U-5 w/o Mo fuel in EBR II, where no cracking has been reported for burn-ups up to 1.2 atoms percent and fuel temperatures up to 566 °C (21).

PuO₂ cermet with high percentage ceramic (75-80 %) have been proposed by BELGONUCLEAIRE for improved core loadings, and current tentative plans are to change to such a core after successful testing of cermet elements in the reactor itself. Such fuels seem well adapted for high power pulsed systems if the volume percent matrix required to hold the element together under high power, pulsed conditions, is low enough to give acceptable critical mass and prompt generation time.

Cladding and core structural material: The choice of cladding and vessel material for SORA is being based on the available experience and data from fast breeder materials programs. BELGONUCLEAIRE has proposed Incoloy-800, but our current thinking is going toward stainless steel. Irradiation effects at our rather low operating tempera-

tures are fairly well known. However, a structural material testing program is planned during the initial operation period to investigate damage and annealing effects.

Coolant: Operational convenience outweighed the advantages of reduced fire hazard and improved heat removal properties in the choice of eutectic NaK as coolant.

Moving reflector: Comparative calculations confirmed that Be would give the highest reactivity worth for the moving reflector. The low density of Be gives important stress and weight advantages for the rotor which carries the moving reflector. The required mechanical properties can be met by beryllium alloys now available. At the operation temperature ($\sim 100^{\circ}\text{C}$), the principal radiation damage effect is hardening and loss of ductility due to helium production. These effects become significant beyond 10^{19} nvt (> 1 MeV) and severe ductility loss has been observed at $3\text{--}6 \times 10^{20}$ nvt (> 1 MeV), although Be has been used in non-stressed applications to well beyond 10^{22} nvt (> 1 MeV)(22). In the SORA moving reflector, the maximum flux is at the front face where stresses are low, and the flux in the region of maximum stresses is about a factor 3 lower. The lifetime of a Be reflector piece has been tentatively set at two years operation, at which time the average integrated fast neutron exposure in the Be block will be about 1.6×10^{20} n/cm² and the peak exposure, 9×10^{20} n/cm² (for 600 kW power). A surveillance and testing program is planned.

Control and safety rods: The use of reflector control rods outside the NaK coolant circuit allowed the choice of materials to be made primarily on physics grounds. For the fast safety rods, Be alloy was specified for its high worth and low weight (calculations indicated a small effect of the added beryllium on the pulse characteristics). Tungsten has been specified for the other control rods for its good reflecting properties and epithermal absorption.

4.2. Reactor Core Design

Thermal-hydraulic design: The SORA core is made up to 123 closely packed fuel elements of 15 mm O.D. and about 60 cm height. Some elements contain Mo in place of fuel and will be used for reactivity adjustment. The elements are supported from the top and are divided into three concentric assemblies for purposes of fuel handling. The NaK coolant enters through a central pipe below the core, is distributed in a lower diffuser

region, flows upward through the core, and leaves the vessel from an upper coolant plenum. Core thermal design data are given in Table 4.1. These data were established under the restriction of 400 °C maximum fuel temperature, set on the basis of available data on U-Mo fuel swelling under irradiation. It is currently being considered as part of the fuel reappraisal to raise the operation temperature enough to continuously anneal out irradiation damage effects to the vessel and cladding.

Preparations for hydraulic tests using water are currently under way in order to determine the coolant flow distribution in the core and to establish the inlet diffuser geometry. The prevention of flow induced assembly vibrations will also be studied as a part of these tests.

Table 4

Thermal Design Data for 600 kW U-Mo Alloy Fueled SORA Core

Core power density	
average	0.10 MW/l
maximum	0.16 MW/l
Fuel element	
average heat flux	50 W/cm ²
maximum heat flux	80 W/cm ²
maximum temperature	400 °C
Coolant conditions	
inlet temperature	200 °C
flow rate	15 liters/sec
velocity	6 m/sec
average temperature rise	60 °C
maximum temperature rise	80 °C
Primary circuit pressure drop	
vessel	1.7 kg/cm ²
total	2.4 kg/cm ²

Thermo-mechanical effects of pulsing: Thermal or mechanical effects of the pulsing on the fuel element, core support structure, pulsation device, or core surroundings are of particular interest in pulsed systems. For SORA, the normal situation in the core can be stated as a 1 °C average fuel temperature rise (for 600 kW and U-Mo fuel) in 50 μsec at a frequency of 50 times per second. Because the rise time is shorter than the time for

propagation of sound waves along the length of the fuel slug, the energy input in the pulse sets up longitudinal vibrations in the fuel and in the cladding. Theoretical studies of these phenomena which are currently under way have shown that for the SORA fuel element about 85 % of the available energy goes into fuel vibrations and about 15 % goes into the cladding (less than 1 % goes into "fuel jumping") (23). The normal 1°C temperature rise causes clad stresses on the order of 10 kg/cm^2 , more than 100 times less than the elastic limit before irradiation. Fuel stresses and strains are also very small. Thus, no problems are expected from the normal operational pulsing.

Considering a single large pulse, present results indicate that damage to the fuel element commences with a fuel temperature rise of about 150°C , which would require an accidental reactivity input of about 80 pcm (see section 5.4). Fracture of the U-Mo fuel after high burnup and plastic strain of the cladding could occur in this range of temperature rises. Such temperature rises would also bring the U-Mo fuel temperature into the phase transition range.

Periodic temperature variations in the fuel, NaK bond, and cladding due to the pulses have also been studied. At the hottest point in the core, the fuel temperature variation in one period of about 1.4°C causes a variation in the average NaK bond of about 0.5°C , and a variation of the inner and outer cladding surface temperatures of about 0.15°C and 0.05°C , respectively. The $50\text{ }\mu\text{sec}$ temperature rise in the fuel is smoothed out very much in the bond and the cladding.

At the present time, an experimental program is being defined to investigate another aspect of thermal pulsing effects: effect of the normal pulsing on the radial NaK bond.

Core vessel and window design: The reactor vessel is a double walled structure having in the core region an irregular hexagonal shape. This shape was dictated by the requirements of high moving reflector worth, minimum distance between fuel and moderators, and high reflector control rod worths, and has been found acceptable due to the low operating pressure in the vessel (about 3 atm. absolute at the core inlet). The space between the double walls serves for leak detection and leak containment. The vessel window between the core and moving reflector is con-

ceived of as a double walled structure with horizontal ribs, welded into the rest of the vessel. The caisson-type structure gives good resistance to deformation with limited material thickness. The total thickness was set at 7 mm with 5 mm material on the basis of calculations; an increase in thickness seems feasible based on critical experiment results and is currently being considered. A program to test window and vessel fabrication techniques and mechanical properties is currently under way.

4.3. Pulsation Device Design

The moving reflector beryllium block dimensions, 24 cm height, 11 cm width, 7 cm thickness, and the speed of movement past the core, 280 m/sec, were set from reactor physics requirements. After very careful examination of possible designs, the present design was developed with the basic philosophy that the mechanical safety features and the operational reliability have to be maximized.

The most important component of the pulsation device is the rotor which carries the beryllium reflector piece. The rotor has a vertical axis of rotation, a radius of 90 cm, and a rotational speed of 3000 rpm. High rotational stability is obtained by a symmetric distribution of the moments of inertia in the horizontal plane. The rotor is a three-armed propeller which is unsymmetric in order to obtain a low reactivity effect of the two compensating arms. The rotor is made of a titanium alloy chosen for high strength to weight ratio and low activation under irradiation. The whole 24 cm rotor, including the Be block, is divided into four identical layers. This is necessary to obtain the required forging treatment of the titanium alloy and at the same time it is desirable because of nuclear safety considerations. A failure of one rotor layer cannot cause a simultaneous failure (within a few microseconds) of another rotor layer.

Each Be block is keyed to a main rotor arm layer by three high strength steel keys. These keys are prestressed by four titanium wedges in such a way that at the test speed of 3600 rpm (20 % overspeed) the mechanical contact of the beryllium block, wedges, and rotor arm is still guaranteed. Stress calculations and photoelastic tests of the fixation by SIEMENS-SCHUCKERTWERKE

confirm that the fixation will stand a test at a speed of 6000 rpm. This test is planned for the first rotor in order to verify the operational safety of the system. Based on the results of photoelastic tests, the safety factor of the rotor (based on overall stress) against fracture is 6 and against yield strength is 5 at operating conditions. Form factors for stress concentrations are not larger than 1.7.

The shaft consists of two conic hollow shaft butts screwed to the rotor. The propeller is supported and guided by the two oil lubricated ball bearings. In addition, there are emergency graphite bearings which act directly on the rotor hub if the rotor is radially displaced by 0.25 mm.

The rotor operates in helium gas at about atmospheric pressure inside a tight aluminium housing. Thereby a Mach number of 0.3 and a rather low friction power of 30 kW is obtained. This value was obtained from model tests performed by SIEMENS. The heat is removed through the finned aluminium housing by a special nitrogen cooling stream which is forced between the housing and an outer casing. The rotor temperature in operation will be less than 100 °C. The energy input to the Be block and rotor during a pulse is so small that no effects on operation are expected.

Special design and testing effort has been given to the window of the pulsation device housing, opposite to the vessel window. Welding tests by SIEMENS proved that 2.5 mm thick forged aluminium window can be welded into the cast aluminium housing under controlled prestress. Further experimental investigations showed that the rotor does not cause gas vibrations which would cause any window deflections.

5. DYNAMICS AND CONTROL

5.1. Kinetic and Dynamic Behaviour

A PFR is more sensitive than a steady state reactor to a change in reactivity level. The PFR follows the same Inhour equation, if a fictitious delayed neutron fraction given approximately by

$$\beta' = \frac{\alpha}{2t_0 E_p} \quad (16)$$

is used (4). $2t_0$ is the time during which the reactivity is above prompt critical. The fictitious β' is not related to the

real β ; its value is quite independent of the fuel used. In SORA, $\beta' = 31$ pcm for U-235 fuel, so that a reactivity change is multiplied by about 20 in pulsed compared to steady state operation. With Pu-239 fuel, $\beta' = 24$ pcm, and the multiplication factor is 8.

Reactivity fluctuations on the time scale of the pulse period produce pulse height changes in a PFR. The fractional change in peak power of a pulse due to a change in reactivity is approximately (4):

$$\frac{\Delta P_m}{P_m} = \frac{2t_0}{\tau} \Delta \epsilon \quad (17)$$

(This relation is accurate only up to 10 - 20 % variations.) Ten percent pulse height variations in SORA are induced by 2.4 and 2.2 pcm reactivity changes for U-235 and Pu-239 fuel, respectively. Thus, acceptable pulse height stability requires reactivity stability to a few pcm.

Such rapid, small fluctuations in reactivity could occur due to mechanical, thermal, or hydraulic effects. Some of the possible sources are: core vibrations, variation in the distance between moving reflector and core, control rod vibrations, changes in moderator properties, and movement of cover gas entrained in the coolant. It is expected that the last four effects can be acceptably limited by good design or operating limitations. (For example, the moving reflector vessel short time distance variation specification is ± 0.015 mm maximum, ± 0.005 mm desired, equal to ± 1.0 and ± 0.3 pcm, respectively, for the present design.) Core vibrations will be studied experimentally as part of the test program currently under way.

On the basis of the above considerations, it was decided that the control system for SORA must be able to counteract small reactivity changes on the time scale of the pulses, i.e. 20 milliseconds. Since this time is short compared to the time constants for heat transfer in the fuel elements, the control feedback loop acts faster than the thermal feedback loops, except those due to the fuel (these prompt effects are discussed in section 5.4). Temperature coefficients for SORA are in any case small; calculated values for the Uranium fueled core are:

Cladding temperature coefficient	- 0.69 pcm/°C
Coolant temperature coefficient	- 0.27 pcm/°C
Fuel axial expansion temperature coefficient	- 0.65 pcm/°C
Doppler coefficient (at 250 °C)	+ 0.16 pcm/°C.

The dynamic behaviour of SORA is thus relatively independent of the power level because of the small temperature coefficients.

5.2. Operational Control System

Startup, operational control, and safety requirements led to the choice of 7 reflector control and safety rods in SORA. The functions of these rods were given in Ref.(3). To repeat briefly, there are two fast safety rods, two rods used principally for start-up, one compensation block (located below the core) for reactivity shimming, and two control rods for pulsed operation. The total reactivity controlled by these rods is calculated to be 3.5-4 %.

For the pulsed operation control system speed of response and safety considerations suggested a fast, rotary rod with a reactivity worth of about 10 pcm for 180 ° rotation and a maximum reactivity insertion rate of 20 pcm/sec, and a vertically moving slow control rod with a reactivity worth of about 40 pcm and a maximum reactivity insertion rate of 1 pcm/sec. The slow control rod is moved step by step to reset the fast control rod into the center of its range.

An analogue study of SORA has been performed in order to investigate the dynamic characteristics including control on the time scale of the pulses (24). The principal conclusions of this study were: satisfactory transient response required a control system time constant of no more than 50 msec, with 20-30 msec preferred; with a maximum reactivity input rate of 20 pcm/sec and an optimized control system gain, the transient due to a step reactivity input of 5 pcm was damped out in about 0.6 seconds; reactivity "noise" with amplitudes up to ± 2 pcm causes no troubles and does not require a control deadband; velocity control using a D.C. motor with large rotor inertia, connected directly to the fast control rod shaft, is a feasible and satisfactory fast control system. Because of the uncertainties in the reactivity variation inputs, the optimization of the actual control system will be possible only after operation of the reactor system.

The control actuation can be based on a signal proportional to the pulse height (peak power) or to the energy in a period. The

design currently calls for using fission chambers which detect 1 msec (halfwidth) thermal neutron pulses produced in the graphite shield.

5.3. Operational Safety System

Operational safety requires that the peak reactivity reached during the reactivity pulse be reduced reliably below prompt critical as rapidly as possible in event of a scram signal. The original specifications for the reactivity insertion rate required 50 mm travel of a safety rod in less than 50 msec.

Assurance that the above performance can be achieved in practice has necessitated testing with a drive system mockup at Ispra. A pneumatic device working with air at 10 atm pressure and a dry friction type braking system has been designed and a mockup constructed. Tests with a proper choice of fast acting valves gave as best performance 46 msec for 50 mm travel (simulated safety rod weight was 14 kgm).

Theoretical calculations for several hydraulic propulsion systems have indicated that reduction of the time for 50 mm travel to less than 20 msec (one pulse period) is feasible. This would mean that a scram initiated by an abnormally high pulse can stop the reactor at the next pulse. A test program has been started on a self-braking system in which the drive cylinder is located between two oil accumulators at different pressures. The system is kept in cocked position by an explosively operated latch, and for scram the latch is released in 1-2 msec. Calculations indicate that with a maximum oil pressure of 70 kg/cm^2 , the initial 50 mm travel will require less than 10 msec.

5.4. Reactivity Excursions

Reactivity excursions have been studied in rather great detail for SORA. The principal purposes of this work were:

- to clarify the importance of fast reactivity feedbacks on power transients,
- to determine the relationship between safety system response speed and allowable accidental reactivity insertion rate,
- to determine the limits on reactivity insertion rates to prevent damage to the core,
- to determine energy releases in cases of transients severe enough to cause fuel vaporization.

Two classes of excursions are to be considered separately:

- class A excursions: those of moderate severity in which

the fuel remains in the solid state and retains its solid state properties,

- class B excursions: those of such great severity that some of the fuel is vaporized and sufficient vapor pressure generated to destroy the system explosively.

For the class A excursions, reactivity feedback occurs predominantly through the Doppler effect and the axial thermal expansion of the fuel slugs. The timescale of an excursion is too short to bring other effects into operation (e.g. thermal expansion of core structural materials) since they depend on the relatively slow diffusion rate of heat from the fuel. The Doppler and axial fuel expansion feedbacks compete against each other since the former is positive and the latter negative. In addition, while the Doppler effect is prompt, the axial fuel expansion is delayed (25). The delay is due to the transit time of longitudinal elastic waves traveling from the end to the center of the fuel slug with a speed of about 3×10^5 cm/sec (thus giving a delay of about 40 μ sec for the SORA fuel element). A computer program, Doppelas, has been written which computes the mean temperature rise in the fuel during the excursion, given the reactivity input, taking into account these competing feedback processes (26).

For the excursions of class B, the temperature rise generated in the fuel goes beyond the melting point of the metal and produces an appreciable vapour pressure. According to Bethe and Tait (27), Jankus (28), and Nicholson (29), the shutdown mechanism in an explosive accident is the movement of core material along the gradient in the vapour pressure from regions of high to regions of lower reactivity worth. The value of this pressure gradient is obtained from an equation of state coupling the vapour pressure of the fuel with the fission energy release. The formalism based on this picture has been built into a computer program (called Sorex-1) which evaluates the explosive energy release in units of kgm TNT (10^6 calories) for any given reactivity input.

The excursions discussed in this section are those during which the pulsation device continues to operate. (Studies on excursion due to failure of the pulsation device are discussed in section 6.) In such cases, the reactivity input is a normal pulse which has been raised to a higher amplitude. This situ-

ation is brought about by an upward drift in the base reactivity of the system. Such a drift will produce a sequence of pulses of increasing amplitude and if the inherent delays in the action of the safety system are too long, one of these enhanced pulses will damage the core. With the present safety system, the first and the second pulses after the start of an accidental ramp reactivity insertion cannot be affected, but the third and later pulses will be controlled.

The programs Doppelas and Sorex-1 have been used to evaluate the consequences of an enhanced pulse. The pulse representation which has been used is a parabolic function of the time around the maximum value ϵ_m (115 pcm in normal operation), and a linear function at the edges with a reactivity insertion rate of 61 sec^{-1} . Results of these calculations are given in Fig. 6 (class A) and Fig. 7 (class B) (26).

The curves for class A excursions indicate that:

- a) the reactivity pulse amplitude at which fuel element damage commences is about 200 pcm, thus 85 pcm above the normal operational value,
- b) the reactivity pulse amplitude at which fuel melting occurs is 223 pcm,
- c) the existing feedback processes have no effect below the fuel damage threshold and only a small influence on the fuel melting threshold.

The consequences of a class B excursion (core destruction) are shown in Fig. 7 in the form of a graph of the mechanical energy release versus the base reactivity drift rate (or the maximum reactivity introduced by the reactivity pulse which causes the excursions, $\epsilon_m = 115 + n \dot{\epsilon}_0$, pcm).

It will be noted that the threshold of destructive excursions lies at $\epsilon_m = 238 \text{ pcm}$ which is only 15 pcm above the threshold of fuel melting and that, for accidental reactivity insertions greater than about 800 pcm, the energy release saturates at a constant value of 105 kgm TNT. This phenomenon occurs because for large reactivity insertion, core disassembly begins before the reactivity has arrived at the parabolic part of the pulse. Thus, the excursion is governed only by the constant ramp rate, $\gamma = 61 \text{ sec}^{-1}$.

From Fig. 7, it is also seen that the reduction of the response time of the safety system to less than 20 msec, by allow-

ing the second enhanced pulse to be controlled, would raise the threshold insertion rate from 52 pcm/period to 118 pcm/period.

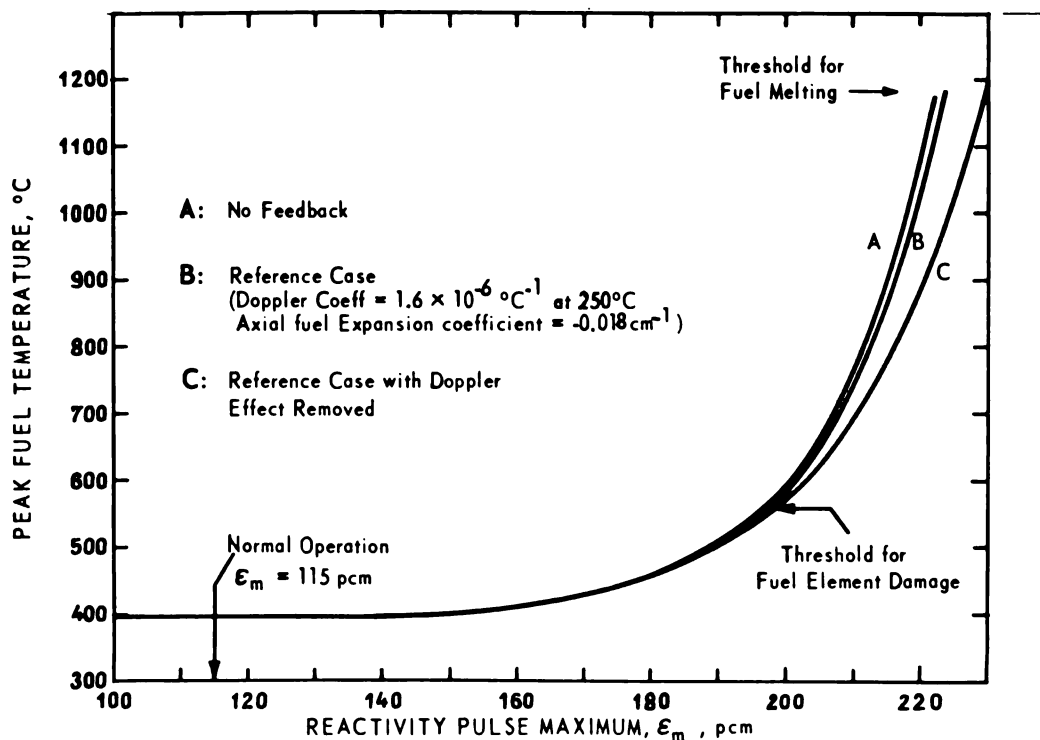


FIG. 6: MAXIMUM FUEL TEMPERATURE REACHED DURING AN ACCIDENTALLY LARGE PULSE (CLASS A)

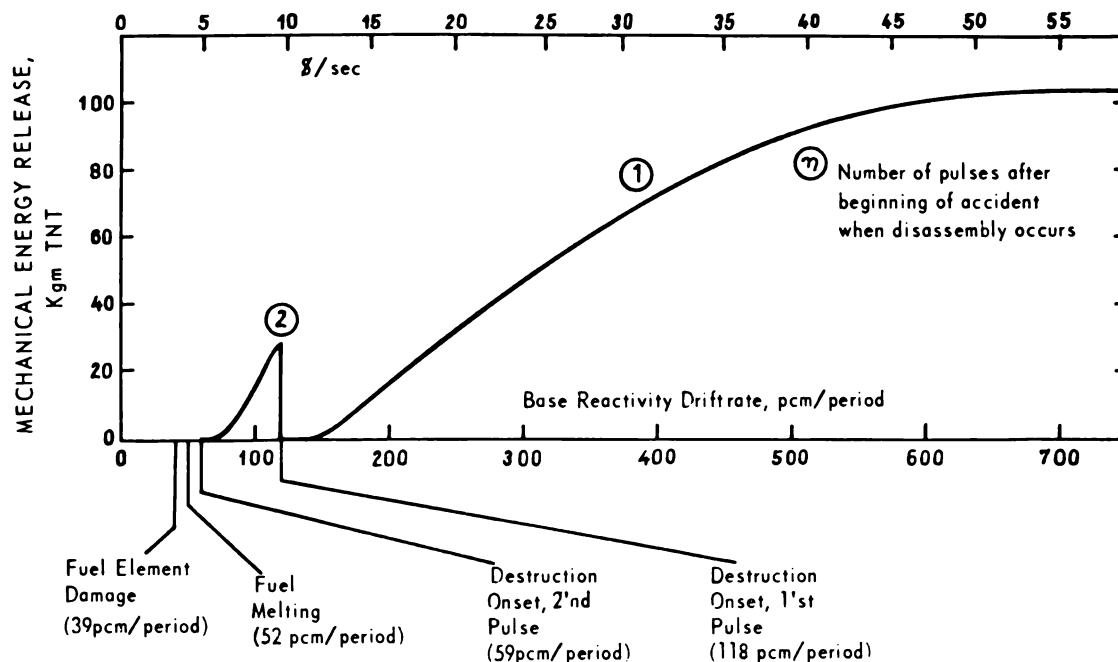


FIG. 7: ENERGY RELEASE AND DAMAGE THRESHOLDS VERSUS ACCIDENTAL DRIFT RATE IN BASE REACTIVITY

6. REACTOR HAZARDS

The SORA design has two principal novel features: the use of a liquid metal coolant in a physics research reactor, and the operation of a high speed rotating device close to a nuclear reactor core. Both of these features have received extensive design and theoretical consideration.

Because of the very rapid reaction of NaK with air, the amount of NaK in the reactor vessel has been kept to a minimum (~50 liters) and most of the radioactive primary NaK circuit and components are located in a special room in the basement of the containment which is kept under an atmosphere inert to NaK. The piping connecting this room to the reactor vessel and the vessel itself are double-walled so that a leak in the inner piping can be detected and contained. In the areas in the reactor block where NaK could spread in case of leakage through the double containment, an inert nitrogen atmosphere is also used. In addition, automatic fire alarm and fighting systems are foreseen.

The concept of using a rapidly rotating device to take the reactivity level periodically above prompt critical sounds hazardous at first glance. Five years successful, and even rather trouble-free, operation of the IBR provide one solid basis for allaying these worries.

An argument often heard questioning the safety of a PFR concerns the sudden stopping of the moving block in front of the reactor before the safety system can operate. The SORA pulsation device has about 7 Megajoules of stored rotational energy; the initial slowdown of speed at 3000 rpm after a loss of power is about 0.2 % per second; the emergency breaking system takes 5 minutes to stop the rotor. The safety system will bring the peak reactivity below prompt criticality in less than 50 msec. It is clear that the postulated situation cannot physically appear.

Speed variations of the pulsation device are also sometimes claimed to be a serious hazard. The theory of PFR's shows that velocity control to 2 - 3 % is sufficient to limit pulse height variations to less than 10 % (4). For SORA. 0.5 % speed control at 3000 rpm is specified.

A possibly more serious question is the position stability of the moving reflector piece relative to the core in SORA. As mentioned in Section 5.1, the allowable rapid motion has been limited in the pulsation device design to give a reactivity variation of less than 1 pcm by careful design of the housing and of the bearings.

In addition, the distance of the outer surface of the Be block from the housing will be measured at three points each revolution by special capacitance probes.

As these considerations should indicate, we have found that the special safety problems of a pulsed fast reactor can be adequately handled by careful detailed design. In this way the operational safety can be made high and it can be maintained high by adequate control and maintenance provisions.

Given that the likelihood of a major failure of the pulsation device is extremely small, the consequences of breakage of a reflector piece or rotor layer could be severe, if the broken piece hit the core. A thorough theoretical study of the consequences of such failures has been underway at Ispra since mid-1964. The methods used are based on the methods developed for other fast reactor excursion analyses, with extensions to describe the mechanical behaviour of the broken part of the pulsation rotor (23). The results give a conservative value for the maximum amount of energy which could be released by a nuclear excursion resulting from a rotor breakage. The models used and data employed in this analysis are periodically subject to improvement or revision, but the results obtained to date show that:

- a) in no case could the fracture of a piece of the Be reflector lead to a destructive nuclear excursion;
- b) in some cases of fracture in one of the titanium rotor layers (about 3 % assuming equal probabilities for all positions and angles), a destructive excursion would result; the maximum energy release is obtained for a fracture position at about $0.3 \times$ radius from the axis and in a small span of angular positions around 14° from the maximum reactivity position;
- c) this maximum energy release is less than 170 kgm equivalent TNT.

The biological shielding surrounding the core and pulsation device is designed as a blast shield to withstand an explosion of 250 kgm TNT and to ensure that the integrity of the containment building is not damaged by such an accident. Special ring flanges are included at the ends of all beam tubes and detector channels to prevent missile generation. The containment building design conservatively takes account of NaK - air reactions.

7. CONCLUSION

A PFR presents a number of novel features not encountered in steady state reactors. The design studies for SORA have shown that, at the 1 MW mean power level, the PFR does not pose particularly difficult or unusual operational or safety problems. Experimental and theoretical physics studies have confirmed the suitability of the reflector pulsing concept used in SORA, and have shown that SORA will be an excellent thermal and cold neutron source for time-of-flight experiments. Control and safety studies indicate that normal operational control can be readily performed and that the safety system can prevent core damage over an acceptably large range of accidental reactivity insertion rates. Studies have also demonstrated that the maximum credible accident can be contained by the reactor block and will not damage the containment system.

These results of the SORA reactor studies give a basis for optimism that the design problems posed by higher power pulsed fission neutron sources can be successfully overcome. Operation of the SORA reactor would provide an important step in the development of more intense pulsed neutron sources.

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A MULTIPLE PULSED TRIGA-TYPE REACTOR FOR NEUTRON BEAM RESEARCH

by

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ABSTRACT

A study has been made of a modification of a thermal reactor to permit its use as a rapidly pulsed source of neutrons for beam research. Certain essential features of the inherent safety of the TRIGA have been retained, but changes in the design permit the reactor system to be pulsed many times per second to peak power levels in the 50 to 100 megawatt region. The average power for 50 pulses/second will be in the range 4 to 10 megawatts. Such a peak power will produce a peak neutron flux at the core end of beam tubes of the order $10^{15} \text{ cm}^{-2}/\text{sec}^{-1}$. The width of the pulse (order of 2 to 3 milliseconds) requires a chopper for some neutron beam work, but this pulse width makes it relatively easy to provide appropriate phasing of the chopper. This thermal reactor has characteristics which make it also well suited for use with a large, cryogenic source to produce cold neutrons.

INTRODUCTION

The standard TRIGA reactor⁽¹⁾ has a large prompt negative temperature coefficient which permits it to be pulsed six to ten times per hour to high instantaneous power levels in the order of thousands of megawatts. While retaining this inherent safety feature, a modification is discussed which will permit the reactor to be pulsed repetitively many times per second to a peak power level in the 100-megawatt region with the power between pulses of the order 0.5 megawatt. It is not sufficient simply to pulse the reactor to a power level of the order 100 Mw by adding a step insertion of positive reactivity and relying on the prompt negative temperature coefficient to limit the peak since the resulting pulse will be quite wide (≈ 0.010 sec) with a "tail" which contains a significant amount of unwanted energy. Rather, it is proposed to shape the pulse through the successive insertions of positive and negative reactivity. The programmed insertion of negative reactivity will have the beneficial effect of strongly perturbing the reactor flux and will result in a substantial clipping of the trailing edge of each pulse. Taking advantage of this feature, one can add a larger amount of positive reactivity than actually needed to produce the desired peak power, relying on the clipping to limit the peak as well as the tail of the pulse. The advantage of adding as much positive reactivity as possible in this manner is to decrease the reactor period and thus reduce the pulse width. It is essential to reduce as much as possible the tail of the pulse (due fundamentally to delayed neutrons) as well as the width of the pulse in order to reduce the average power. The insertion of negative reactivity, as proposed here, serves both these needs.

For slow neutron research, the low power of the reactor between pulses will contribute to improved signal-to-noise ratio compared with the use of a reactor operating at a steady state power equivalent to the average of the pulsed reactor. Further improvement of the signal-to-noise ratio is possible without loss of neutron beam intensity through the use of a D_2O reflector tank and tangential beam tubes. Such a system has been incorporated lately in a light-water reactor⁽²⁾ with significantly improved results.

The pulse width at half-height will probably be on the order of 2 to 3 milliseconds. This is considerably wider than desired for some neutron beam

research where pulse widths of the order 20 μ sec are desired though much broader pulses of the order 2000 μ sec may be equally useful for structure studies of single crystals using Laue patterns⁽³⁾. When narrow pulses are needed, it will be relatively easy to phase a chopper system to select maximum beam intensity and provide the desired pulse width. The pulsed thermal reactor considered here also provides an excellent possibility to produce "cold" neutrons in a cryogenic source since the cold source can be as large as desired. The cold source can thus be designed for maximum intensity of "cold" neutrons. In these and similar uses, the proposed system is suitable for beam research and has all the flexibility, inherent safety and economy of a thermal reactor system.

DESCRIPTION OF PULSING SYSTEM

The reactor system proposed is to be much like a normal TRIGA reactor⁽¹⁾ except that:

1. The fuel elements may be slightly modified,
2. Forced water cooling will replace the use of natural convective cooling, and
3. A new pulsing mechanism which provides for sequenced insertion of positive and negative reactivity will replace the usual pulsed control rods.

A rough conceptual sketch of the core and one possible pulsing mechanism consisting of one or two wheels is shown in Figure 1. The core and pulsing mechanism will be located in a usual TRIGA shield with light water surrounding the core. Presumably the wheel(s) will spin in a water-free environment, though this detail has not yet been settled.

Within the general framework discussed above, a number of parameters for the pulsed system can be varied to satisfy various needs. Among these parameters will be notably average power and pulse frequency. Although other selections can be made, this study has been aimed at providing a suitable system with an average power in the range of 4 to 10 megawatts with a pulse rate of 50 per second. While this study does not provide an optimum design of a multiply pulsed TRIGA-type reactor, enough calculations have been made to indicate the usefulness of the concept. Further optimization of the heat transfer properties of the core could allow the average core temperatures to be significantly lowered. This would allow the peak power (and hence neutron flux) per pulse to be increased with no deterioration in

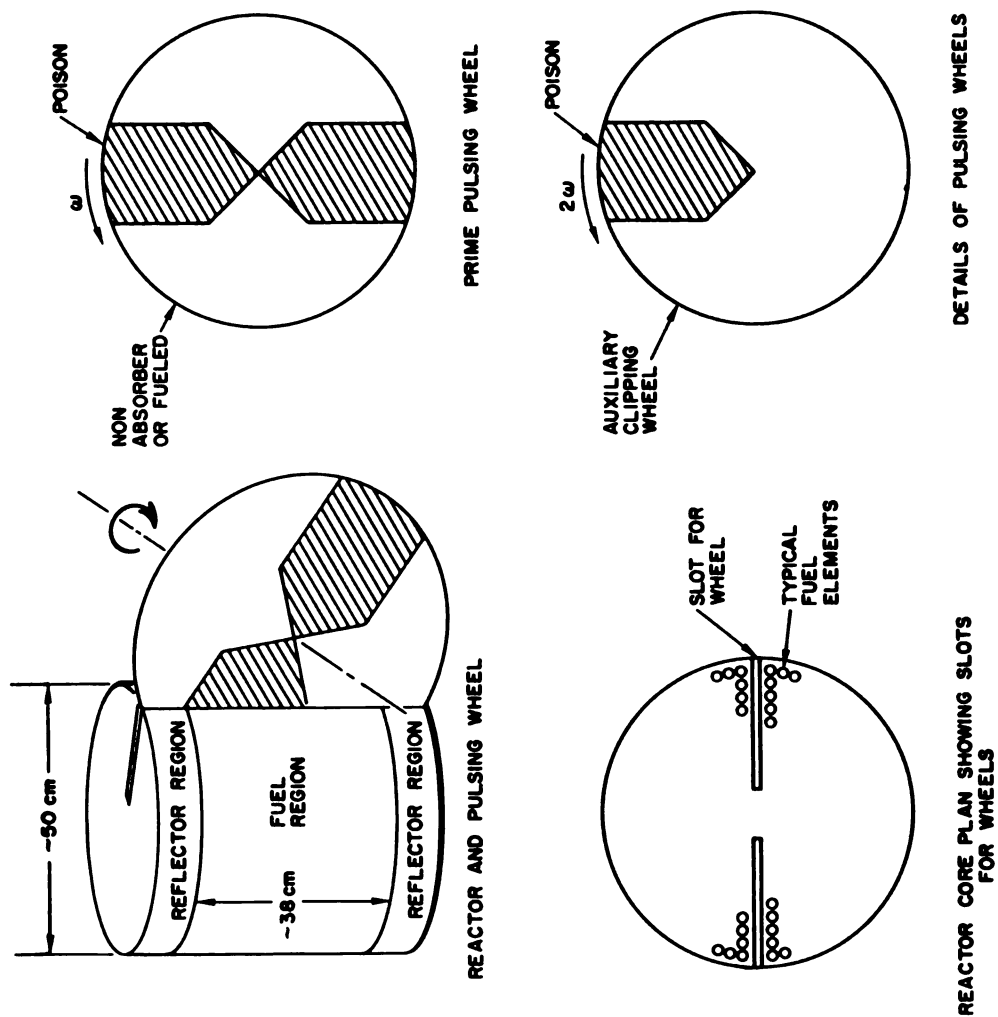


Figure 1--A sketch showing the relation of a pulsing wheel and core. The arrangement of poison and non-poison is also shown for the prime pulsing wheel and a possible auxiliary "clipping" wheel.

the safety considerations for the system. Table 1 is a summary of the results of two different sets of calculations made with the use of the General Atomic kinetics code BLOOST⁽⁴⁾.

REACTIVITY SEQUENCE

The schedule for reactivity (positive and negative) insertions needed to provide the results of Table 1 is shown in Figure 2. The reactor is brought to a steady-state power with the insertion of \$3.00 excess reactivity (above cold critical). The average core fuel temperature is about 365°C. For one schedule of pulsing (shown by the solid curve of Figure 2 and corresponding to the uniform rate of insertion of +\$5.50 reactivity followed by the uniform rate of insertion of -\$6.00 reactivity) a peak power of 40 megawatts is produced. If the insertion of negative reactivity is accelerated as shown by the dotted curve in Figure 2, the narrower pulse results in a 43 percent reduction in average power while the peak power is reduced by only 28 percent. Figure 3 shows the actual time dependence of the equilibrium pulses associated with these two sequences of reactivity insertions.

Additional calculations have shown that 6 kg of U^{235} fuel in the form of a slab placed in a radial slot (similar to the wheel slot shown in Figure 1) will produce a positive excess reactivity up to \$14.85. Further calculations have indicated that sufficient heat will be produced in this fuel insert to require that uranium carbide be used since its melting point is above the calculated temperature rise. A related calculation was made to evaluate the reactivity worth of a B^{10} poison in the same core slot; its worth was found to be \$13.90.

With the above values of reactivity worth for fuel and poison in a radial slot in the core, the actual insertion of positive and negative reactivity can be achieved in a number of manners. A rotating wheel could introduce alternately fuel and poison into the core. It would be easily possible to achieve a fueled section worth \$2.50 and a poison section worth -\$9.00. A difficulty with this would be the heating in the fueled section with the attendant problem of heat dissipation especially if the wheel rotates in a gas enclosed volume. A satisfactory alternative would be to use a rotating wheel containing alternate sections of poison and non-poison. The poison section would need to be worth -\$11.50 which is possible as discussed

Table 1
SUMMARY OF OPERATIONAL CHARACTERISTICS FOR
AN UNOPTIMIZED SYSTEM

	<u>Uniform Pulsing and Clipping</u>	<u>Accelerated Clipping</u>
Pulses per Second	50	50
Peak Power (Mw)	40	29
Average Power (Mw)	7	4
Minimum Power between Pulses	0.6	0.5
Pulse Width at half power (sec)	3×10^{-3}	2×10^{-3}
Energy in Pulse, Mw-sec	0.14	0.080
Peak Thermal Leakage Flux in reflector (n/cm^2 - sec)	$.5 \times 10^{15}$	$.36 \times 10^{15}$
Prompt Neutron Lifetime, μsec	12	12
Prompt Negative Temperature Coefficient ($\delta k/k/^\circ\text{C}$)	6×10^{-5}	6×10^{-5}
Average Fuel Temperature in Core ($^\circ\text{C}$)	365	355
Temperature Variation in Fuel during Pulse ($^\circ\text{C}$)	~1	~1

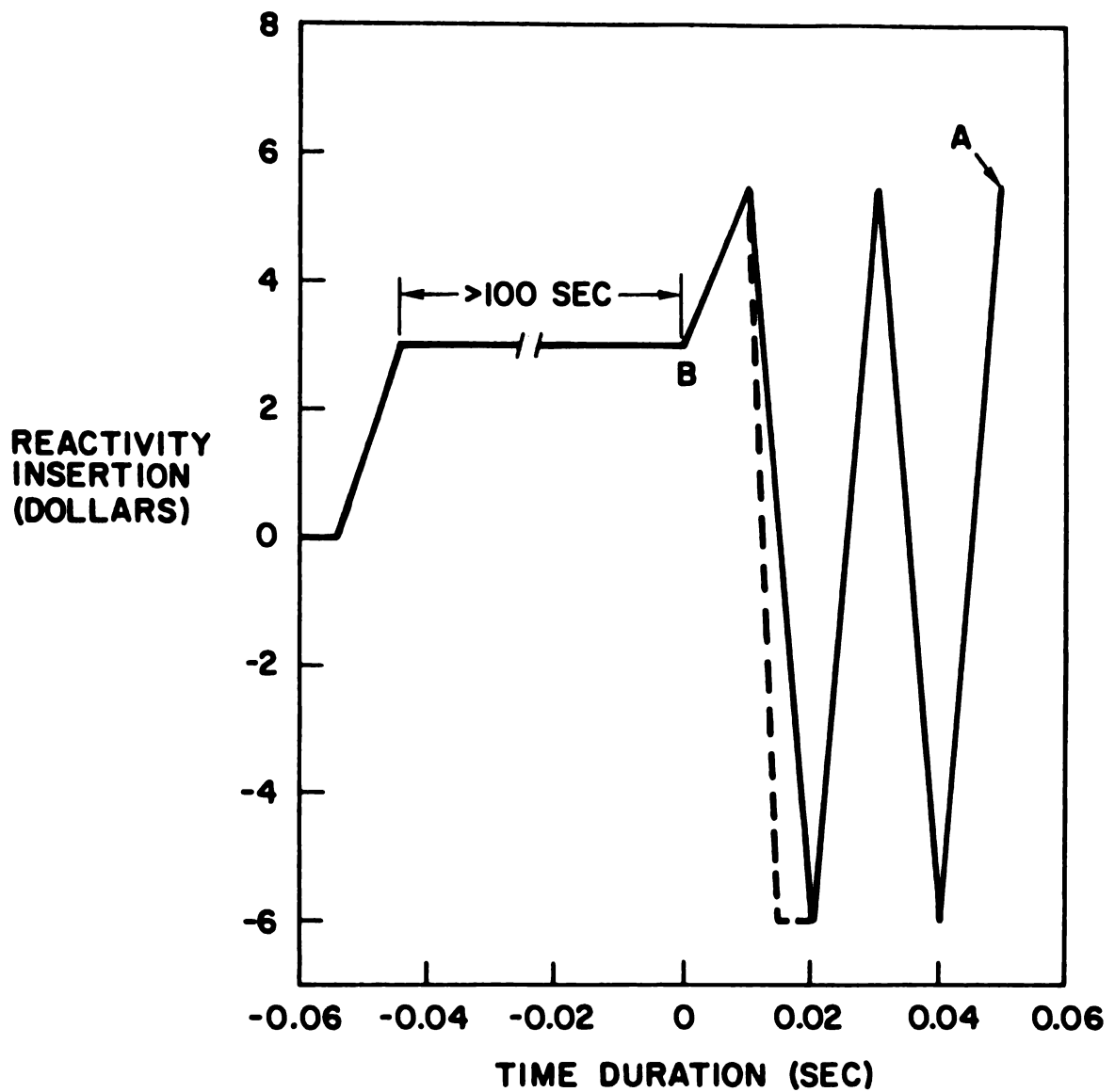


Figure 2--A sketch showing the time dependence of reactivity insertions. Normally the oscillatory behavior is continued indefinitely. For accident analysis, the insertion is assumed to be halted at point A.

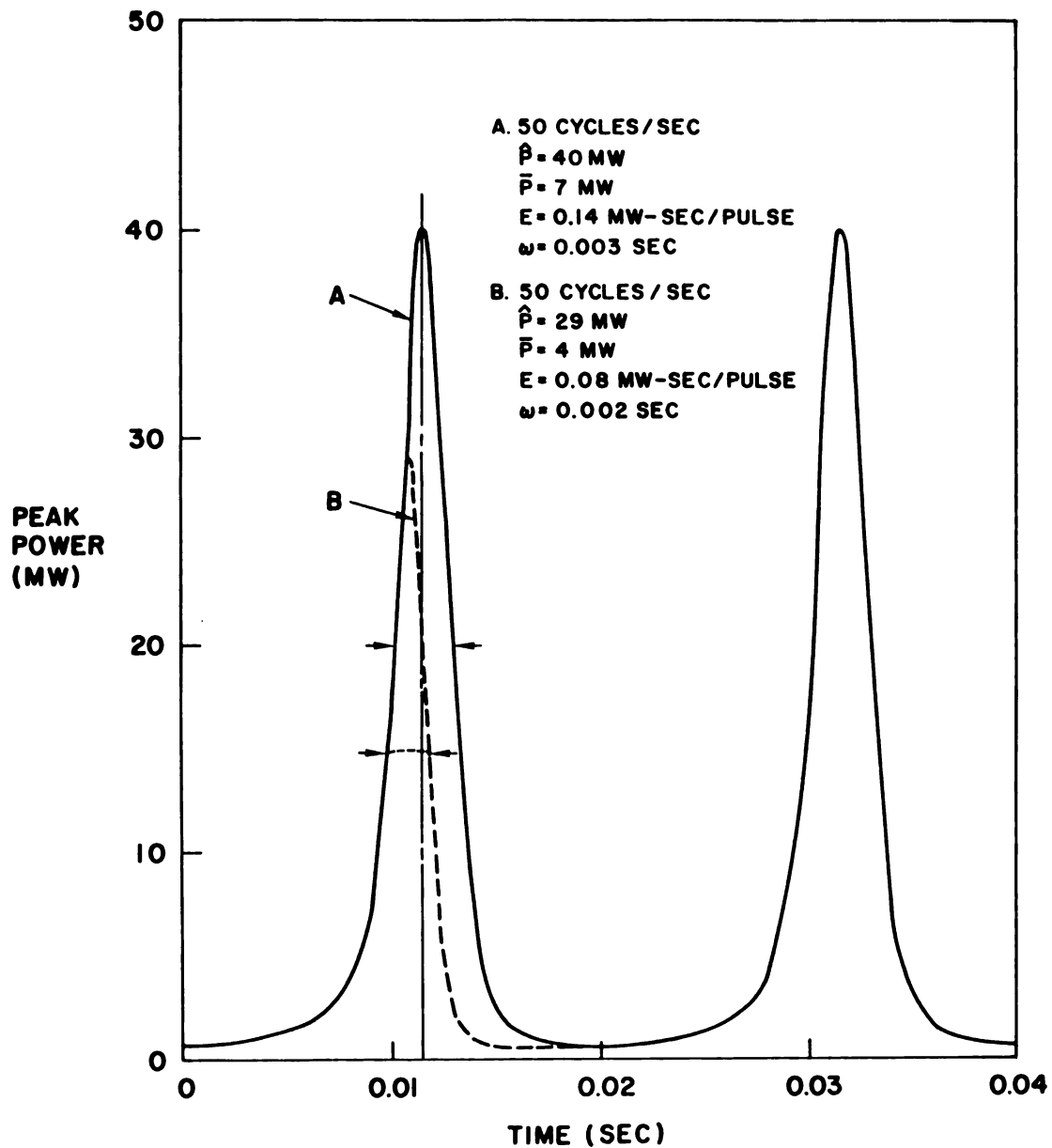


Figure 3--The time dependence of the equilibrium pulses for a uniform rate of inserting positive and negative reactivity. The dotted curve results if the insertion of negative reactivity is accelerated.

above. In this case, the reactor would have +\$5.50 reactivity inserted with regular control rods. The rotation of the wheel would then produce the reactivity sequence shown by the solid curve in Figure 2.

If it is desired to clip the trailing edge of each pulse by accelerating the insertion of negative reactivity, this could be accomplished by the use of a second wheel with poison located as shown in Figure 1 and rotated at twice the speed of the prime pulsing wheel. By properly phasing the two wheels, it will be possible to insert an additional amount of poison into the core during that portion of the pulse when the prime poison is being inserted. No change would take place during the portion of the pulse when the prime poison is being withdrawn.

An additional benefit accrues from the postulated use of a wheel which provides two pulses for each revolution. For 50 pulses per second, its rotational speed would be 25 rps or 1500 rpm, a relatively low speed. There should be no difficulty in obtaining the necessary mechanical strength of the rotating members which might have a diameter as large as 60 cm. Of course, an auxiliary wheel of about the same size used as described above would rotate at twice this speed.

REACTOR CHARACTERISTICS

This pulsed system will require some changes in the design of a standard TRIGA reactor. The change in the pulsing mechanism has been discussed above. The heat removal characteristics of the core must be modified to allow for the higher average core power of the proposed system. For the calculations reported in Table 1, the heat removal rate from the fuel was increased by a factor of 2 and the water coolant flow increased by a factor of approximately 10, compared with the usual natural convective cooling. These changes are readily achieved in a practical design. While considering temperature effects, it is worthwhile noting that the temperature coefficient does not provide the shutdown mechanism on a time scale suitable for the rapid pulsing rate. Before the temperature rise becomes significant for this purpose, a shutdown mechanism is provided by the mechanical addition of negative reactivity. The prompt negative temperature coefficient is still available as an inherent safety feature. The prompt neutron life-time is assumed to be 12 μ sec and the prompt negative temperature coefficient is assumed to be $6 \times 10^{-5} \frac{\Delta k/k}{^\circ\text{C}}$. These are factors of 2 or

3 smaller than the parameters of a standard TRIGA reactor and are computed characteristics of a core composed of modified, more highly loaded, smaller-diameter fuel elements.

SAFETY CONSIDERATIONS

Safety of the proposed system has been partially demonstrated by the performance of standard pulsed TRIGA reactors in which single reactivity insertions of +\$5.00 have been safely and repeatedly made in a cold core. The resulting peak energy release of about 1.0 megawatt-sec per fuel element caused no indication of fuel element distress. In additional pulsing tests of fuel elements, the calculated peak temperature in some elements have safely reached as high as 1170°C. Since the hydrogen-to-zirconium ratio in one of these elements was about 1.65, a fuel-moderator element having a lower hydrogen-to-zirconium ratio of 1.58 to 1.60 can safely reach peak temperatures of ~1250°C.

Since calculations have agreed well with the observed reactor parameters for the pulsed standard TRIGA, it is reasonable to place confidence on the same calculational techniques to compute the pulsing characteristics of the proposed system for a maximum credible reactivity insertion. The worst accident for the various system of pulsing proposed above occurs when \$5.50 positive reactivity is inserted in the warm (365°C) core (point A in Figure 2) and no subsequent negative reactivity is provided. Such a case would occur if the rotating wheel were to stop rather abruptly or to fly out of the core. A transient analysis indicates that the peak power and energy released in such an event would be 11,300 megawatt and 52 megawatt-sec, respectively, with a resulting momentary peak temperature in the hottest fuel element of about 1250°C, which falls to a lower steady state temperature. With a hydrogen-to-zirconium ratio of 1.58-1.6 for the proposed system, the equilibrium pressure corresponding to a peak temperature of 1250°C for the maximum credible reactivity insertion is no higher than already experienced in deliberate pulse tests.

Appropriate design parameters, including the hydrogen-to-zirconium ratio and the heat transfer characteristics of the core, can thus be selected to assure safety for the reactor system in this maximum contingency.

SUMMARY

Calculations have verified that a thermal reactor of the TRIGA type using U-ZrH fuel elements and light water coolant can be operated to give routinely many pulses per second. Peak pulses in the range 50 to 100 megawatts and peak thermal neutron^{leakage} fluxes of $\sim 10^{15} \text{ cm}^{-2}/\text{sec}^{-1}$ can be achieved. Pulsing can be accomplished with a wheel with relative ease since there is no requirement on precise mechanical location of the wheel within the core slot. It is envisioned that the proposed system would be constructed within a normal, above-ground concrete shield and would appear quite similar to a normal TRIGA except that the fuel elements would be of smaller diameter and forced water cooling would be required. To achieve the best signal-to-noise ratio for some neutron experiments as discussed earlier, a small tank of heavy water would probably replace a part of the light water core reflector and be used to feed the core end of tangential beam tubes. In addition, a cryogenic source of "cold" neutrons would be incorporated as desired for experiments. Using this overall concept, it becomes possible to produce a competitive intense source of neutrons for research purposes, using a reactor which possesses important elements of inherent safety and economy.

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DISCUSSION
OF
PAPERS III.A.1 (Larrimore) and III.A.2. (Whittemore)

Chairman: V. Raievski

Secretary: T. Wimett

MICHAUDON: In order to improve the resolution for resonance neutrons, that is up to 1 keV, one might install a 1- μ sec chopper in phase with the reactor and close to it. In that case, what would be the relevant loss intensity in a given solid angle?

RAIEVSKI: I want to point out that for neutron energies in the range of 1 eV or more, as was said by Larrimore, we are planning to couple a linac which would decrease the pulse width down to 6 μ sec.

KLEY: As a matter of fact, we have analyzed the situation also using a fast chopper in connection with the pulsed reactor. Of course, you know the pulsed reactor produces a burst of 50 μ sec so you cannot use all your neutrons. With a chopper you may reduce the pulse to a limiting burst width of, say, $1/4 \mu$ sec. Now, you lose, of course, up to 200 times as many neutrons as are used depending on the initial burst which you are shaping with your fast chopper. You still can compare the available intensity after you have chopped it, for instance, with a linac of 40 kW and 1- μ sec burst time (as you have, for instance, at the RPI accelerator). If you do this, you find that the fast chopper arrangement at the pulsed reactor is still about six times better than the linac which gives you pulses of 1 μ sec. This is, of course, only true for neutrons up to about 1 keV where there is also slowing down time in moderators. Beyond about 1 keV, you would not use any moderators at the linac and you could reduce inherently the pulse width and the accelerator would be much better than the pulsed reactor. So you find a certain range -and this is only true again for small samples- where the chopper in phase with the pulsed reactor is better than the linac. There is development of linacs in the nsec range, but no more power than about 5 kW has been achieved to date. There are projects to arrive at 50 kW for nsec pulsing techniques, and of course future linacs would be much better. Still, if we compare it with existing linac facilities having about 10-nsec pulses and an average power of 5 kW which corresponds to about the best linac we have available in Europe; then we find that the SORA reactor is still about six times better up to 1 keV.

MICHAUDON: If you install a chopper at a reactor, the loss of intensity is not just the ratio of the pulse widths because of the area of the slits. Did you

include the loss of intensity because of the small area of the slits, or did you assume that you took the case of a small sample?

KLEY: We have calculated the detection rate for the appropriate slow and fast detectors. We have considered a high efficiency detection system for the fast chopper system operated at the pulsed reactor, because the inherent pulse is long, allowing us to have detectors with higher efficiency. At linacs with 10-nsec bursts, you must use very fast detection systems. So we have considered what is possible on the detection side using available techniques for the linac as well as for the fast chopper with a pulsed reactor and have concluded that the latter system is six times better.

MAIER-LEIBNITZ: We now have three reactors with a thermal flux of 10^{15} , high flux reactors with continuous flux, and two types with about 50 short pulses per second. One of these has a pulse duration which may be directly useful for a number of applications. The other one requires additional equipment to select energy or pulse duration. This situation shows the necessity of detailed comparison of experiments that may more favorably be done with one or another of these types. Would one of the proponents like to indicate a field where his reactor is most applicable?

RAJEVSKI: I believe that you may have better ideas of how to use these reactors than the speakers. In any case, this will be discussed in the Panels. Before asking the speaker who is waiting to answer this question let me make a small comment on these 10^{15} neutron fluxes. We have seen a value of 0.5×10^{15} for the TRIGA reactor. And for the SORA reactor, you have given the figure 1.9×10^{15} with a factor of 2 for the geometry which would give something like 4×10^{15} so you see that the ratio of the two is of the order of 8. It's 10^{15} but with a very large span. With regard to a 50- μ sec pulse, this is the most conventional pulse width which is used in steady-state reactors, for instance, and it will be discussed at the Panels. With regard to a 5-msec pulse, this is less known and I wonder for which kind of experiment Whittemore believes a 5-msec pulse would be especially interesting?

WHITTEMORE: There isn't much I can say that would be new on this point. If 50 pulses per second is a reasonable pulse frequency in order to do a fairly large class of beam experiments, then the reactor that I have discussed here would come close to satisfying that need if a separate chopper is used to provide the narrow pulse. If one wants to make an alternative supposition that perhaps one should start by designing the experiment completely first, and arrive then at some different number for pulse frequency as reasonable and necessary, then of course this particular system will not have been optimized for those conditions. Rather I have taken the condition that 50 pulses per second seems like a reasonable number taking into account resolutions in energy and momentum, and reasonable flight paths and

detector sizes in the present state of the art. Under these circumstances 50 pulses per second does not seem particularly incorrect. I must admit that this has not been optimized at all.

KLEY: The pulse characteristic of your reactor really is remarkable, but I did not see clearly a detailed design of the wheels. You have a very large core and I'm wondering if you can really arrange wheel designs that provide enough protection for your bearings? We are slightly in a more favored condition on SORA; we have a very small core, and are able to protect our bearings from radiation damage by geometric arrangement.

WHITEMORE: We have not made a complete detailed design of the wheel or its bearing. I don't see why it should not be possible to construct rather easily with the relatively modest diameter that we're proposing. It should also be possible to get the hub of the wheel sufficiently far from the core to protect it against unreasonable radiation damage.

SHAFTMAN: Why do you use the clipping wheel just to get down to your reactivity shutdown margin faster rather than to increase your shutdown margin?

WHITEMORE: It's a matter of how many dollars of reactivity you can get in a given size of blade -how much poison- and how much you have to move. It's a matter of practical design. I don't see any reason why you couldn't increase the magnitude to something like the maximum that we showed you can get, namely 14 dollars, but in one of the designs that I talked about and seem to favor, I already require 11.5 dollars to shut the reactor down, so I don't see how I could get much more in one slot because it's already completely black to thermal neutrons. You can always do any of those more complicated things such as use more slots and more wheels. Now you're talking definitely of using two wheels, I have tried to use just one wheel.

MAIER-LEIBNITZ: I want to stress again that the planners of a reactor should look a little more into this question of what it could be used for. It would seem to me from something I have learned yesterday from Wilkinson of Oak Ridge, that it would seem appropriate to propose a use of this msec pulse reactor for the investigation of organic single crystals. Here it would be as good as any other reactor except for this factor of two or four in flux compared to normal high flux reactors. I do not think that one should always wait for experimenters to say what they want in an experiment and then say we can change our flux by a factor of two. It is you who should look a little more into the question of how such a pulse can be altered for this application.

BURAS: I would like to say that after reading the paper and listening to the presentation, I do not think that SORA is an underdeveloped advanced project. How-

ever, I am concerned about the repetition frequency. With a repetition frequency of 50 cps you have a time between two pulses of 20 msec. You intend to use a cold neutron source and a thermal neutron source which means that long flight paths will be used. For, let us say, 8 Å with a velocity of 500 meters/sec this will mean that, in order to avoid overlap of pulses, you are limited to a maximum pulse flight of 10 meters. If you compare this with the existing IBR reactor with 5 pulses/sec, you are limited for the same figure to a flight path of 100 meters, and, of course, with the same width, maximum resolution on SORA will be less by a factor of ten. This is something that will worry everybody who would like to work with cold neutrons or even thermal neutrons.

RAJEVSKI: I believe that this is the kind of comment which is very important for the reactor designer -- to know what pulse widths and repetition rates are required. Again, this would be a topic for the panel discussion. I agree that reducing the pulse frequency is advantageous for high resolution experiments, but there we must face one problem -- that reducing the repetition rate means you are increasing the thermal cycling and thermal stresses on the fuel elements. In the present SORA design we have attempted to be conservative and avoid this thermal stress. At the IBR condition of 5 pulses/sec the increase of temperature in the fuel element would be of the order of 12 degrees. This may be acceptable, but we have no experience in this area. The way which is foreseen to avoid reducing the repetition rate is by coupling with an accelerator. It would be easier for SORA than using two wheels.

AGERON: What are the thermal, epithermal and fast fluxes at the nose of the beam tubes between the pulses? Secondly, what are the size and composition of the cold source?

LARRIMORE: As regards the power or flux between pulses, one can get this very easily from the paper, but this is very conservative. For the peak-to-background, my optimistic number for our final condition is a ratio of 6500 for the fast flux. The calculations show that for the thermal source, this peak-to-background ratio will be about a factor of two lower. Therefore, the background thermal flux to peak thermal flux will be in the order of 3000. However, you can do other things. You have the possibility, of course, of shutting off your beam tubes during the 20 msec between pulses. If you do this with a rotating collimator, you can get this background down much further. Besides this, the use of neutron tubes and such things presents the possibility of reducing these backgrounds even more. The pulse reactor definitely has a major advantage, just how much is something perhaps someone else would like to comment on. Regarding the cold source, you need more thickness since you have to moderate the neutrons as well as to rethermalize. In the paper we show that if you have a reasonable moderator density in your cold source, allowing for cooling channels of maybe 80% (perhaps that's a little optimistic), you will need something

like an equivalent slab of between 5 and 7 cm depending on what you want. Our moderator holes are 14 cm in diameter into which you will be able to install cryostats. You have almost this much diameter available for ice or whatever other material can be used. We have considered having a half-cylinder of ice and a back half-cylinder of beryllium if you want very cold neutrons. If you don't want the very cold neutrons below the beryllium cutoff, you may not use the beryllium filter, but you may do something else. The design of cold sources is in a rather early state for our reactor. There are all sorts of things which are beginning to be considered; about how you arrange the cooling, which is not too high actually as you can see in the paper, i.e., do you put it preferentially in one place, toward the back away from the core?

AGERON: The question concerned not exactly the ratio between the flux in the pulse and flux in the background, but, for instance, the ratio of thermal flux to epithermal flux in the pulse, or in the background; is it the same?

RAIEVSKI: I believe that if you refer to the paper of Larrimore, you would find some numbers for these ratios which are characteristic of the source geometry. Making a very large source, you have a smaller epithermal flux and a large thermal flux. Conversely, a small source tends to harden the spectrum. There is no one answer for your question, but you will find numbers in the report.

KLEY: If I understood your question, you asked for the ratio of the thermal flux to the fast flux during the peak and the thermal flux to the fast flux during the time in between the peaks. This ratio is, of course, different because in the time between pulses we have a steady-state situation and not a time dependent situation. Between pulses we do have, as a matter of fact, a different ratio of thermal-to-fast flux than during the power peak because during the power peak the thermal flux shape does not follow the fast flux, so we have less thermal flux buildup, by about a factor of 3 as I remember.

BECKURTS: Do you believe that SORA as now designed could be operated at a considerably higher power? What are the limitations in power?

RAIEVSKI: You are anticipating the presentation of Hoffman this afternoon, so we should not go too much into detail.

LARRIMORE: The initial power planned is 600 kW. With plutonium oxide cermet fuel we could go to 1 MW. At this point, I don't want to predict what the operation people will want to do in 1975 or 1980. Our heat exchangers are now designed for 1 MW. I don't think that it will be possible to go much beyond 1 MW without fairly sizeable changes, for instance, in the reactor vessel. This is not impossible because, as you will hear later on, with these sorts of systems, you have to plan to be able to replace practically everything and it is not inconceivable that when we

go to plutonium we would be able to use a thicker vessel, change geometry, and in the same system it could be possible to go up another factor of two in power.

MILTON: I wonder why the SORA designers have their rotating reflector in such a way that if it flies apart, it flies into the reactor. Why not use a horizontal axis so that if it flies apart it just goes into the floor or the ceiling?

RAIEVSKI: It's a very interesting question, and we have been bothered with this. There are many reasons.

LARRIMORE: There are three axis directions about which you can rotate. The question is whether any of them doesn't have a component toward the core? The answer is that, when we got around to trying to get a design for the bearings, we found that even the one which makes breaking pieces go out to the floor or the ceiling can't be totally in this direction, but has to have a small tilt. We therefore got into a position where we were comparing a horrible looking design tilted and bent in some strange angle, where the component toward the core might be some 10% of the velocity, versus a very nice geometry design such as we now have with vertical shafts. This has a much higher safety factor for a number of reasons. We chose operational safety against some possible small improvement in ultimate safety. One possibility is to avoid two bearings and use an overhanging wheel. There also are ways which you can make things look a little better, but for our system we did consider all these things at length. If you look at the safety factors which we quote, I think you must come to the conclusion that the possibility of having a failure of this device is extremely small, and, as we've shown, even if it does fail, it's not going to affect the population, and hopefully not even the experimenters.

SHAFTMAN: I'd like to make a comment about a point made by Maier-Leibnitz. It seems to be a very rare situation where reactor designers are competent to judge what the users would want. Typically, reactor designers work with users who specify what they want. Very often it is not possible to tailor the reactor design to satisfy certain particular uses, but I had hoped that one good thing that would come out of this meeting would be much better specifications of what users want so that reactor designers can attempt to meet these specifications.

RAIEVSKI: This is what we are expecting from this symposium.

THE SUPERBOOSTER

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SUMMARY: A design study has been made at A.E.R.E. to determine the suitability of accelerator-booster systems for the provision of thermal neutron beams for solid state research. The system considered was a pulsed travelling wave electron linac, of 300 kW mean, 200 MW peak beam power operating into a subcritical plutonium fuelled liquid metal cooled fast reactor. The reactivity of this assembly was oscillated in phase with the linac pulsing so that the multiplication of delayed neutrons emitted between pulses is reduced while maintaining maximum usable multiplication at the moment of the pulse.

Problems considered include design of the accelerator and beam handling system; design of the reactor core; stresses set up in the fuel due to thermal pulsing; methods of oscillating reactivity; thermalisation of neutrons; layout of beam holes and number of neutron beams available; design of the containment building and general safety problems.

The conclusion reached is that it should be possible to build a plutonium-carbide-fuelled reacting system operating at a prompt multiplication during the pulse of 300 and with a mean power of around 9 MW. This could feed up to 20 neutron beams (various geometries have been considered) from an effective thermal flux during the pulse of 2×10^{16} n/cm² sec. The pulse length and repetition rate for this are ~ 60 μ sec and 200 pulses/sec, respectively. Alternatively, it is possible to use smaller moderators to yield a pulse of ~ 20 μ sec long, when the effective flux is reduced to 7×10^{15} n/cm² sec.

1. INTRODUCTION

This paper gives the results of a design study made at Harwell to assess the possibility of producing intense pulsed thermal neutron beams to be used for solid state research. The method proposed is a combination of accelerator and subcritical multiplying assembly (booster). This principle is not new; the present Harwell Booster has been in operation since 1958 and consists of an electron linear accelerator (45 MeV, 0.5 amps peak, 5 nsec to 1.7 μ sec pulse length, 500 pps max) feeding into a subcritical U²³⁵ fuelled fast neutron multiplying assembly which has a central source multiplication of approximately 10. Fast neutrons leaking from its surface are partially moderated by thin slabs of water, and the epithermal (0.1 eV upwards)

neutrons emitted from these slabs feed neutron beams for nuclear experiments. This facility was primarily designed for use in high resolution neutron spectroscopy and an important design parameter was the shortest neutron pulse length attainable. When making a time-of-flight experiment an increase of neutron intensity at the expense of an increase in neutron pulse length only leads to the necessity of using longer flight paths with a corresponding reduction in intensity at the detector. It can easily be shown that for a constant detector size an n-fold increase in intensity is required to compensate for an n-fold increase in pulse length. It is thus necessary to examine carefully the time in which the multiplication of fast neutrons takes place and to relate this to the shortest usable pulse length for a given experiment. This latter is determined by two factors, viz., the time it takes the fast neutrons to moderate into the energy region under observation and the availability of multichannel time analysers with this resolution. The moderation time to any given velocity is roughly given by the relation

$$\tau = \frac{4}{V}$$

where V is the neutron velocity in centimetres per second and τ the moderation time in seconds. Typically this time is ~ 3 μsec at 1 eV, 0.3 μsec at 100 eV and ~ 0.1 μsec at 1 keV. At the time of commissioning of the booster (1958) the shortest channel widths provided by easily available multichannel time analysers were around $1/4$ μsec and it was estimated that this would not be improved to better than $1/10$ th μsec within five years. In the event, this surmise proved to be correct. It was therefore decided to design the booster for a fast neutron rise time of $1/10$ μsec . The choice of fuel was restricted to U^{235} for safety considerations, so that as the lifetime of a fission neutron in pure U^{235} metal is 5.8 nsec, a maximum prompt multiplication of 10 was possible, allowing for a possible increase to 10 nsec in neutron lifetime due to dilution of the fuel in a practical booster.

The maximum multiplication usable is also restricted by the need to keep the background arising from the delayed neutrons down to a manageable value and, using U^{235} , a multiplication of 10 proved to be suitable from this point of view also. Had plutonium, with its lower delayed neutron fraction and with its shorter neutron lifetime, been available, then this figure could have been doubled.

The neutron booster designed to these specifications was described at the Second Geneva Conference on the Peaceful Uses of Atomic Energy (1) and has been in constant use since that time. It has proved to be an extremely trouble-free piece of equipment, the only problem arising in use being caused by erosion of the electron beam window by the mercury coolant. It is now normally driven by electron pulses 0.1 μsec long at a repetition frequency of 200 pps, and feeds neutrons to about 10 beams. These are used for neutron time-of-flight experiments with flight paths ranging from 5 to 300 metres in length.

Due to the use of relatively short pulse lengths, and so small amounts of moderator, the booster as it stands is inefficient as a producer of thermal neutrons and it is instructive to look at the differences to be expected in a booster required for thermal neutron work. As thermal neutrons are required, the moderators must now be larger and the time to moderate to thermal will be considerably greater than that to moderate into the electron volt region. Under these conditions the neutron pulse shape is determined not only by the moderation time but also by the effective lifetime of thermal neutrons in the moderator and a compromise has to be struck between efficient moderation and reasonably short lifetime. The argument that in use the shortest possible neutron pulse is required, holds for thermal neutron experiments as much as for higher energies. Later in the paper it will be shown that pulse lengths from ~ 20 μsec to ~ 80 μsec are typical in thermal neutron moderators of the size that might be used to feed a beam experiment. Thus it becomes possible to increase intensity by increasing the accelerator pulse length from $\sim 1/10$ μsec up to as much as 20 μsec , and it is in addition possible to increase the multiplication up to the point that the reactor lifetime is around 10-20 μsec .

Before considering the design of a multiplying target to use with an electron linear accelerator, it is relevant to see what thermal and epithermal fluxes could be obtained from the use of accelerator and moderator alone. The largest accelerator suitable for neutron production which it was considered could at the moment be routinely designed has the specification shown in Table I. (As a result of the design study it was found that this rating was low by perhaps a factor 2.) Using this accelerator into roughly optimised water and heavy water moderators, both mean and peak fluxes have been estimated, and these are shown in Table II together with the expected mean pulse length. It will be seen that from the point of view of peak neutron intensity, only ordinary water need be considered as a moderator. High mean fluxes obtainable in heavy water are only obtained in consequence of the very long thermal neutron lifetime and do not correspond to high peak fluxes. It will also be seen that the fluxes obtained are not in any way competitive with the fluxes obtained from a high flux reactor, and as the cost of such an accelerator plus associated building would be in the region of two million pounds, it would not be a very attractive proposition for the production of thermal beams.

2. THE USE OF A BOOSTER FOR THE PRODUCTION OF THERMAL NEUTRONS

It has already been seen in the introduction that the long thermalisation time for neutrons allows relaxation of the limitation on multiplication imposed by the need for a fast pulse rise-time. However, if we impose the condition that the booster must at all times be subcritical, then assuming delayed neutron fractions of .62%, .23%, and .19%, respectively, there is an overall limitation of the fast multiplication to 160 for U^{235} , 480 for U^{233} and 520 for Pu^{239} . Further discussion is limited to U^{235} and Pu^{239} fuels. It is equally clear that at any given multipli-

cation less than the maximum permitted, Pu^{239} will give a significantly lower delayed neutron background than will U^{235} . This delayed neutron background has two undesirable features: firstly it produces a steady between-pulse background for all experiments and secondly it increases the mean power of the device, and so the engineering difficulties, by a significant factor. This last point is extremely important to the production of the highest thermal neutron flux, as in order to make the best possible use of the neutrons it is necessary to use the smallest possible core size to obtain maximum fast flux to feed the moderators. Increase in core size merely to dissipate the extra power caused by delayed neutrons is therefore highly undesirable. The percentage of power usefully employed for various multiplications is shown in Table III. From this it can be seen that useful multiplication can only be achieved if some means is adopted to reduce the effect of delayed neutrons. Accordingly, it is proposed to modulate the reactivity of the booster in phase with the pulses from the accelerator by a mechanical control arm or control wheel so that the booster is at maximum reactivity at the moment when the electron pulse arrives, and the reactivity is reduced to as great an extent possible between pulses, thus reducing the multiplication during the 'delayed neutron regime'. In order to assess the effect of such modulation a sinusoidal law has been assumed and Table III also shows the percentage of useful power for several amplitudes of modulation. It can be seen that at the higher multiplications (e.g., $M = 300$ for plutonium) even a $\pm 1\%$ oscillation will cause a very significant increase in the fraction of power usefully employed. A further feature of the reactivity oscillation is that it allows an increase in the maximum multiplication that can be employed as it is now possible to let the prompt multiplication rise above $1/\beta$ (β = delayed neutron fraction) without the reactor becoming overall supercritical. The superbooster therefore is a subcritical multiplying assembly driven by an electron linac in which the reactivity of the assembly is modulated in phase with the pulsing from the linac so as to maintain the highest possible prompt multiplication at the moment of the pulse, and to minimise the steady background produced by delayed neutrons.

To determine the relation between multiplication and pulse rise time for a practical booster it is necessary to establish a core composition. This is determined by engineering considerations and the need for a minimum core size to maximise the flux. These considerations dictate the highest possible core density, and so high fuel temperatures, and for this reason plutonium metal cores are unattractive. For the purposes of the design study it was established that either plutonium carbide or plutonium oxide fuel in a sodium cooled core would be feasible from the engineering standpoint, but that the plutonium carbide core would have somewhat better pulse and flux characteristic. Later studies were therefore confined to the use of a plutonium carbide fuelled system. The core composition reached is given in Table IV which also

shows the mean reactor neutron lifetimes for that composition. In determining the reactor neutron lifetime it is assumed that the core is reflected by approximately 10 cm of nickel. This thickness is a compromise between minimum critical mass and acceptable neutron lifetime in the reactor. From this it is apparent that a lifetime of about 8 μ sec can be achieved in a practical carbide fuelled core for multiplications of up to 300. With a sinusoidal reactivity oscillation of $\pm \sim 1\text{--}1/2\%$, the mean operating power will be around 9 MW for 300 kW of electron beam power. The principal parameters of this core are included in Table IV.

Two methods have been considered for modulating the reactivity. Initially it was considered that 3 rotors would be mounted adjacent to the core, each comprising a nickel cylinder with its axis vertical and with a wide diametral slot cut into it in the manner of a 'Fermi' neutron chopper. Mockup experiments using the VERA critical assembly machine showed that the amount of reactivity which could be controlled by these rotors was small, and as they occupy a considerable portion of the core surface (and reduced by a factor of at least two the number of beams that could be extracted from the unobstructed core) this method of modulation is not a practical one. The alternative is to follow the design suggested for the SORA reactor. This comprises a portion of reflector mounted upon a large diameter (2 metres) wheel rotating about a vertical axis, which approaches the core at the moment the electron pulse is fired from the accelerator. A mockup of this system has been made for the SORA project at the Oak Ridge National Laboratory and shown to control approximately 7 β reactivity in a U^{235} fuelled core. It is thought that such a wheel could be built without occupying more than 25% of the available core surface. The use of a large diameter wheel also simplifies the engineering of the bearings by removing these to a greater distance, so enabling shielding to be interposed between them and the core. The layout of the reactor is shown in Fig. 1. It consists of a hexagonal target region, 8 cm across on the vertical axis of the core. This is surrounded by an annular core 20 cm o.d. containing about 35 kg of PuC fuel in the form of clusters of fuel pins whose axes run parallel to the axis of the core. The core is mounted in a pressure vessel through which liquid sodium is circulated. The electron beam enters the target from above.

This core is surrounded by a nickel reflector approximately 10 cm thick and the moderators to produce the neutron beams are embedded in this reflector. These moderators are discussed in detail in the following section. Part of the reflector must be cut away to accommodate the rotor. The complete reactor is mounted inside a massive biological shield through which pass the neutron flight paths closed at the outer ends by pressure tight windows. Access to the core is obtained through the top of the biological shield.

3. THERMALISATION OF THE NEUTRONS AND DESIGN OF THE MODERATORS

Ideally one would like to produce a pulse of thermal neutrons which follows the shape of the fast neutron pulse without broadening. However, as has already been mentioned, this is not possible and the whole reactor design is a compromise based upon the properties of moderators. As fast thermalisation is required, it is clear that only hydrogenous moderators need be considered for this reactor, and the design of moderator has been made on this basis. No particular choice has been made between water and organics and this choice can be made on the basis of engineering considerations. Some thought has been given to the use of cold moderators to produce higher intensities of low energy neutrons, and if these were to be adopted it would probably be necessary to use liquid hydrogen (although there is some possibility that methane could be used).

Calculations of the energy and time dependence of the neutron flux have been made on various moderator geometries and computer codes. Orthodox reactor physics calculations have been made using CRAM, which is a diffusion theory code in either spherical or two-dimensional geometry. The YIFTAH-B cross section set, extended down to thermal by using cross-sections quoted in ANL 5800 and BNL 325, has been used. Compilers attached to CRAM give power distributions, mean neutron lifetimes and perturbation integrals for reactivity calculations. The time dependence of the neutron spectrum in the moderator can be obtained from the code SPECTIM which solves the time dependent diffusion equation, describing the moderator geometry by means of a buckling. Time average neutron spectra have been obtained from the computer code SIMPH which solves the infinite medium neutron balance equation, also representing the geometry by a buckling. The use of diffusion theory in the spectrum calculations in such small systems is open to criticism as also is the description of the moderator by a single buckling mode. However, diffusion theory with a single buckling mode is probably adequate to calculate the shape of the pulses as a function of neutron energy provided the buckling is chosen to give the correct decay of the total thermal flux. The procedure has been to use CRAM to calculate the mean thermal neutron flux in the moderator box for a given core and moderator box geometry and for a given mean power. SIMPH is then used to calculate the average neutron spectrum and SPECTIM to calculate the time dependence of the spectrum for a δ -function fast pulse. This time dependence is then combined with the real time dependence of the fast pulse to obtain a final time dependence of the thermal spectrum. These three calculations are then normalised together to give the absolute flux as a function of time and energy. From this flux, pulse widths have been calculated for various energies. The results are shown in Table V and the neutron spectrum at peak intensity is shown in Figs. 2, 3, and 4.

In the moderator geometry two approaches can be taken. Either a moderator can be chosen to give maximum possible flux into the maximum number of beams, with no

consideration at all of the pulse length, the pulse being subsequently tailored by the use of a neutron chopper to suit the experiment, or the moderator boxes can be made small in an attempt to keep the pulse length down to a value which can be used directly. In this case, neither is the flux obtained likely to be the maximum nor will the number of beams that can be extracted be as large as in the former case. The advantage lies in avoiding restricting the beam through a narrow chopper aperture. The calculations made have covered both approaches. Buckling values of 0.59 cm^{-2} and 1.448 cm^{-2} correspond to the "small" moderator box philosophy, while the buckling of 0.24 cm^{-2} corresponds to an annular ring of moderator all round the core, which is typical of the first approach.

To prevent thermal neutrons returning to the core lengthening the fast neutron pulse it has been found necessary to incorporate a shield of B^{10} between the moderator and the core, and the effect of this on the flux has been included in the calculations. Examinations of the table will show that a reactor power of around 9 MW, which is considered feasible from the engineering point of view, a thermal flux of 2×10^{16} is possible in pulses around 80 μsec , and it will be seen later that up to 20 beams between 4 and 6 in. in diameter can be extracted from this moderator. The spectrum of neutrons in this case is shown in Fig. 2. This spectrum is considerably harder than what would be expected from a reactor and has a very high epithermal tail (with approximately E^{-1} spectrum). Spectra obtained from the small moderator boxes are shown in Figs. 3 and 4 and are even harder than that shown in Fig. 2. For these spectra the E^{-1} "tail" gives fluxes comparable to that obtained in the peak of the Maxwellian component from a hot moderator at any temperature that can be specified (this arises because for a given total flux, the flux at a peak of a Maxwellian is proportional to T^{-1}). Such room temperature moderator geometries then give neutron spectra suitable for any class of experiment except those requiring very low energy neutrons. Calculations on a moderator at liquid hydrogen temperatures were made for the case of small geometry and it appeared that the peak flux was not increased by cooling the moderator. Although the mean flux was increased, this was almost entirely due to a lengthening of the pulse. Subsequent calculations of a larger moderator show that in this case there was some increase of the peak flux.

Calculations on the room temperature moderator were made using a kernel appropriate to a perfect gas at liquid nitrogen temperatures. The use of more realistic kernels was not considered necessary for these approximate calculations.

4. LAYOUT OF THE PROJECT

An overall layout of the accelerator and reactor is shown in Fig. 5. The accelerator, which is 17-1/2 metres long is mounted horizontally at high level and arranged so that the beam can be directed down into the reactor by means of a 90° bending system. The use of a downwards rather than upwards beam was chosen in

order to minimise the number of breaches in the reactor containment and so improve safety. The reactor is contained in an approximately circular building of diameter 120 feet and is shown mounted off centre in order to give a greater range of possible flight path lengths inside the building. If flight paths greater than 30 metres in length are required these have to penetrate the building, and a subsidiary building is needed to house the detectors. For a quadrant of detectors of considerable radius this subsidiary building could either be correspondingly large or could consist of a central block house with a crescent shaped building around this.

At the time this building was designed the booster was being thought of as having three or four small moderator boxes each serving two beams only, these boxes being chosen to give short pulses. The alternative approach of one large moderator leads to a beam tube layout as shown in Fig. 6, in which up to 21 tubes view the same moderator. With this layout the number of experiments that can be accommodated simultaneously is limited not by the reactor design but by the ability to accommodate the experiments (and the shielding required to reduce the background to an acceptable level) in the space around the reactor. Careful thought must therefore be given when deciding the actual details of these beam holes as to whether better layouts can be obtained by offsetting beams or by using radial beams. A combination of both moderator designs would be possible, a feasible compromise being to have a large room temperature water moderator covering almost 180° of the core surface, together with one cold moderator, covering possibly a further 90° , leaving the remaining 90° for the rotor. These considerations have not been worked out in detail.

5. SAFETY

5.1 Introduction

It is apparent that the superbooster is a potentially hazardous device and consequently its construction and operation would only be approved if it could be shown that engineered safeguards, plus administrative safeguards, could provide adequate safety margins; or alternatively, that if a dangerous mishap occurred adequate containment could be provided to deal with the consequences. The necessary safety study would be a substantial exercise in itself, and would be conducted by an independent specialist group working in close touch with the design team. The designers and the safety group would, in practice, react upon each other to produce modifications in design and safety philosophy, so it is premature at this stage to do more than to point out the order of magnitude of the safety problem which might have to be faced. This is in many ways similar to that for a conventional fast reactor and the solution will therefore tend to be the same as for this class of reactor.

Whenever it can be shown beyond all doubt that the core will not under any circumstances, however unlikely, collapse into a prompt critical assembly, then a

large part of the safety problem vanishes. However, this is not possible for the superbooster and it has been necessary to proceed by evaluating the magnitude of the explosion that would occur in the worst imaginable circumstances; i.e., if the core collapsed to its maximum density, and to consider whether the shield and building can be designed to contain wholly the effects of this explosion. Such a melt-down could start by loss of coolant, by sudden reactivity addition as for example might happen if the rotor were to burst, or by progressive blockage of the subassembly inlets causing melt-down in one fuel element. The use of a doubly-walled vessel around the core with coolant introduction through pipes at the bottom would make it unlikely that the coolant would be lost completely, but it is not certain whether this concept is compatible with the thin window required to enable proper operation of a reactivity pulsing device. The reactivity rotor and the control system would be designed so that if there was any failure during the operation, short of complete disintegration of the rotor, there would be no sudden reactivity addition to the system.

For a spherical model, collapse of the core to maximum density would increase the reactivity by approximately 19%. This figure is uncertain as information is lacking on the density change occurring when plutonium carbide melts. The amount of reactivity controlled by the reflector is in the order of 40%, so to restore the system to a sub-critical state would involve removal of up to 50% of the reflector.

5.2 Energy Release in a Prompt Excursion

Any calculation of the energy release depends on the model used to represent the core in its collapsed state. In this case the threshold model has been considered, in which it is assumed that a certain fixed amount of energy has to be generated before bulk expansion of the core takes place. This energy is the amount required to raise the temperature to the point where the core has expanded and filled up all the internal voids which are present after the collapse, these voids being assumed to be dispersed uniformly throughout the core. It is considered that this model over-estimates the time delay before the core expands and thus over-estimates the actual energy release. Ignoring the effect of the Doppler coefficient and assuming that a reactivity increase of 100 dollars per second is feasible, it has been calculated that the energy released would be equivalent to around 60 lbs. of TNT. An estimate of a Doppler coefficient of reactivity is 6×10^{-6} per centigrade degree at the working temperature and somewhat less than this at the temperature of the molten fuel. An upper limit of 2×10^{-6} appears to be reasonable at this temperature. The effect of this would be to increase the energy release at this rate of addition of reactivity to around 100 lbs. of TNT.

5.3 Model Explosion Experiment

A quarter scale experiment was carried out to investigate the effects of an explosion inside a container of the expected shape of the superbooster biological shield and beam tubes. This experiment indicated that the general pressure level in the chamber immediately after the explosion and before mass flow in the beam tubes has become stabilized, is about 600 lbs. per square inch. In the beam tubes a shock wave of peak pressure 150 lbs. per square inch is first propagated down the tube, followed by a rarefaction and by a gradual pressure rise to 200 lbs. per square inch as the main flow becomes established. Outside the end of the beam tube the shock wave expands and the over-pressure at a point 30 ft away in the reactor building along the axis of the tube is estimated to be about 3 lbs. per square inch. This pressure is approximately doubled at the wall by reflection. The initial smaller wave is followed by an expansion of the main gas flow which generates a vortex ring which will react on loose equipment in the vicinity.

5.4 Containment of a Nuclear Excursion

Assuming that the design of the biological shield is strong enough to withstand the effects of a nuclear excursion the remaining problem is to contain the sodium, which would be expelled down the flight tube. The possibility of containing this by windows over the flight tube was examined and it was found that although a 2 mm steel window could contain these pressures without undue attenuation of the neutrons, it has not yet been possible to establish that these could not be fractured by missiles travelling down the tube. It was therefore necessary to consider the effect of the burning of the sodium (vapour or liquid) that might be expelled from the tubes and to design a building to withstand any pressure rises. With a 100' diameter building the worst conditions assumed (instantaneous burning of all the sodium) lead to a 15 psi shock lasting 6 msec at the wall of the building followed by a steady pressure rise of 0.7 psi max. A preliminary design indicated that a building could be constructed to contain this with adequate margins of safety.

5.5 Conclusions

1. Many of the safety problems are similar to those occurring in any fast reactor of comparable power and the known solutions can be applied.
2. The presence of the flight tubes penetrating the biological shield result in additional problems connected with the ultimate safety of the system.
3. It is not possible to guarantee that the core will not collapse into a prompt critical state due to high temperatures. An estimate of the energy release in such an accident leads to an explosion equivalent to the explosion of 100 lbs. of TNT. This is believed to be an over-estimate.
4. The biological shield will contain the nuclear explosion, but sodium could be

ejected rapidly down the flight tubes. Windows capable of transmitting 80% of the neutrons can prevent the escape of sodium provided they are not fractured by fragments of the core, etc.

5. Preliminary indications are that a containment building and ventilation plant can be designed, which would provide adequate safety margins in the event of the maximum credible accident.

TABLE I ACCELERATOR SPECIFICATION	
Type of Accelerator	L-Band Travelling Wave Linear Electron Accelerator
Accelerating length	17-1/2 metres
Number of sections	9
Type of R.F. drive	Klystron amplifiers
Peak R.F. power	270 MW
Peak beam power	150 MW
Mean beam power	300 kW
Beam pulse lengths	2 μ sec to 20 μ sec
Beam p.r.f.	Variable to a maximum determined by pulse length and mean power available.

TABLE II NEUTRON OUTPUT FROM ACCELERATOR ALONE		
Fast neutrons	7.5×10^{17} n/sec peak	
	1.5×10^{15} n/sec mean	
Moderator	H_2O	D_2O
Thermal neutrons (mean)	7.5×10^{12} n/cm ²	1.5×10^{12} n/cm ² sec
Thermal neutrons (peak)	2.5×10^{14} n/cm ² sec	1.5×10^{12} n/cm ² sec
Mean pulse length	~ 150 sec	∞

TABLE III		
% Modulation	Multiplication	% Useful Power
0	100	0.78
0	200	0.56
0	300	0.33
2%	300	0.6
10%	300	0.72

TABLE IV
PARAMETERS OF BOOSTER CORE

1. Composition	PuC 45% Zr 22-1/2% Na 32-1/2%	
2. Critical mass	25 kg	
3. Rod diameter	2 cm	
4. Coolant	Na or NaK	
5. Reflector	Ni	
6. Heat rating (max)	~350 W/g ~4.5 MW/litre	
7. Surface heat flux	~650 W/cm ²	
8. Coolant inlet temperature	250°C	
9. Coolant temperature rise	100°C	
10. Doppler coefficient	$+6 \times 10^{-6}$	
11. Expansion coefficient	-4×10^{-6}	
12. Long term overall temperature coefficient	-15.3×10^{-6}	
13. Void coefficient (Na)	-2.7×10^{-4} per % reduction in Na density	
14. Neutron lifetime	15 cm reflector	35 nsec
	10 cm reflector	26 nsec
	5 cm reflector	16 nsec
	(PuO ₂ fuel, 15 cm reflector 42 nsec)	

TABLE V, Part 1
NEUTRON FLUX AND PULSE WIDTH

Electron Pulse (μsec)	prf (max avail.)	Multipli- cation	Reactor Power at 200 pps (MW)	Moderator Temp. (°K)	Moderator Buckling (cm^{-2})	Peak Flux in Thermal Group (sub-cadmium) ($\times 10^{15}$)	Peak Flux per eV at 0.316 eV ($\times 10^{16}$)	Pulse Width at Half Height (μsec)			
								0.01 eV	0.0316 eV	0.1 eV	0.316 eV
2	1000	100	0.84	293	0.24	0.64	0.18	49	42	13.5	5
					0.59	0.73	0.21	39	24	11	5
					1.448	0.64	0.27	24	20	9	5
				70	0.24	0.59	0.16	--	--	----	---
					0.59	0.69	0.22	54	20	8	5
					1.448	0.42	0.15	28	15	8	5
5	400	100	2.1	293	0.24	1.5	0.37	50	42	14.5	6
					0.59	1.8	0.44	29	24	11.5	6
					1.448	1.6	0.56	24	20	10	6
				70	0.24	1.5	0.33	--	--	----	---
					0.59	1.7	0.46	54	21	9	6
					1.448	1.0	0.29	28	15	8	6
5	400	200	1.9 to 4.0*	293	0.24	2.9	0.54	53	47	22	8.5
					0.59	3.2	0.64	32	28	16.5	8
					1.448	2.7	0.82	27	24	14	8
				70	0.24	2.7	0.48	--	--	----	---
					0.59	3.0	0.66	55	23	11	8
					1.448	1.7	0.43	31	17	10	8
10	200	200	3.8 to 6.0*	293	0.24	5.8	0.84	54	48	25	11.5
					0.59	6.2	0.98	33	29	18	11
					1.448	5.2	1.24	27	24	16	11
				70	0.24	5.2	0.75	--	--	----	---
					0.59	5.8	1.02	62	24	13	11.5
					1.448	3.2	0.65	31	19	12	11.5
10	200	300	7	293	0.24	8.0	1.06	58	54	32	13
					0.59	8.4	1.25	37	33	22.5	12.5
					1.448	7.0	1.57	30	28	20	12.5
				70	0.24	7.2	0.93	--	--	----	---
					0.59	7.8	1.30	58	26	15	12.5
					1.448	4.2	0.83	34	21	14	12.5
20	100	200	3.8 to 6* (at 100 pps)	293	0.24	11.0	1.02	56	52	32	20
					0.59	11.4	1.19	36	33	25	20
					1.448	9.2	1.49	30	28	23	20
				70	0.24	9.7	0.92	--	--	----	---
					0.59	10.4	1.24	57	27	20	20
					1.448	5.3	0.78	33	23	20	20
20	100	300	7.2 (at 100 pps)	293	0.24	15.6	1.43	60	56	38	21
					0.59	15.8	1.68	39	36	28	20
					1.448	12.7	2.10	33	31	26	21
				70	0.24	13.6	1.29	--	--	----	---
					0.59	14.4	1.75	67	30	21.5	20.5
					1.448	7.2	----	36	25	21	----

* Depending on whether or not reactivity oscillation is used.

TABLE V - PART 2
Neutron Flux and Pulse Width for Annular Moderator

Electron pulse	p.r.f.. (max. available)	multiplication	reactor power at 100 p.p.s.	Peak flux in thermal group	Peak flux per eV at .316 eV	Pulse width at half height (μ s) (estimated)			
						.01 eV	.0316 eV	.1 eV	.316 eV
20 μ s	100	300	7.2 MW	2.6×10^{16}	2.4×10^{16}	55	50	30	21
10 μ s	200	300	7 MW	1.4×10^{16}	1.8×10^{16}	53	48	28	14
20 μ s	100	200	~ 4 MW	1.8×10^{16}	1.7×10^{16}	53	48	30	20
10 μ s	200	200	~ 4 MW	1.0×10^{16}	1.5×10^{16}	50	44	22	11

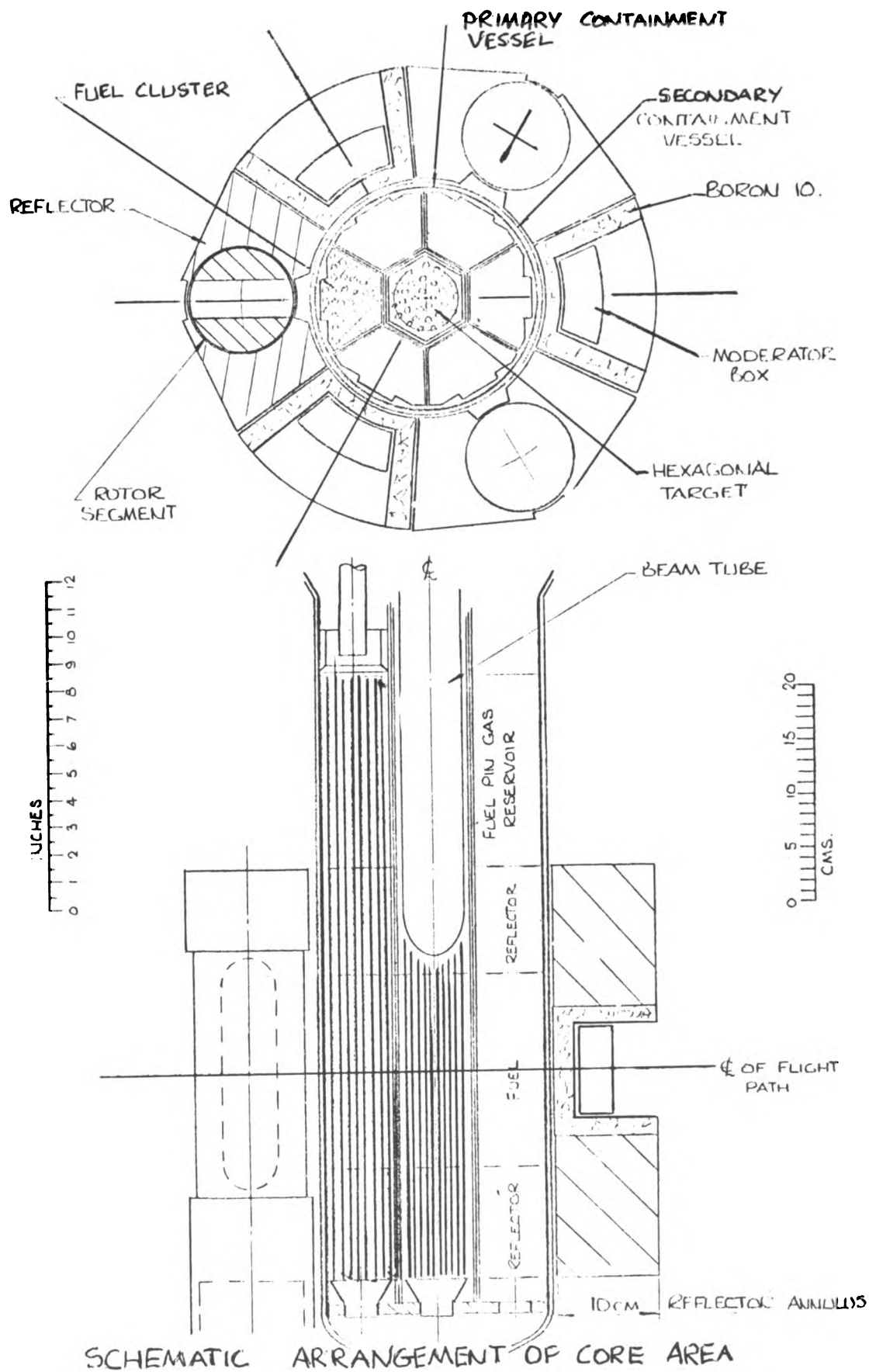


Fig. 1

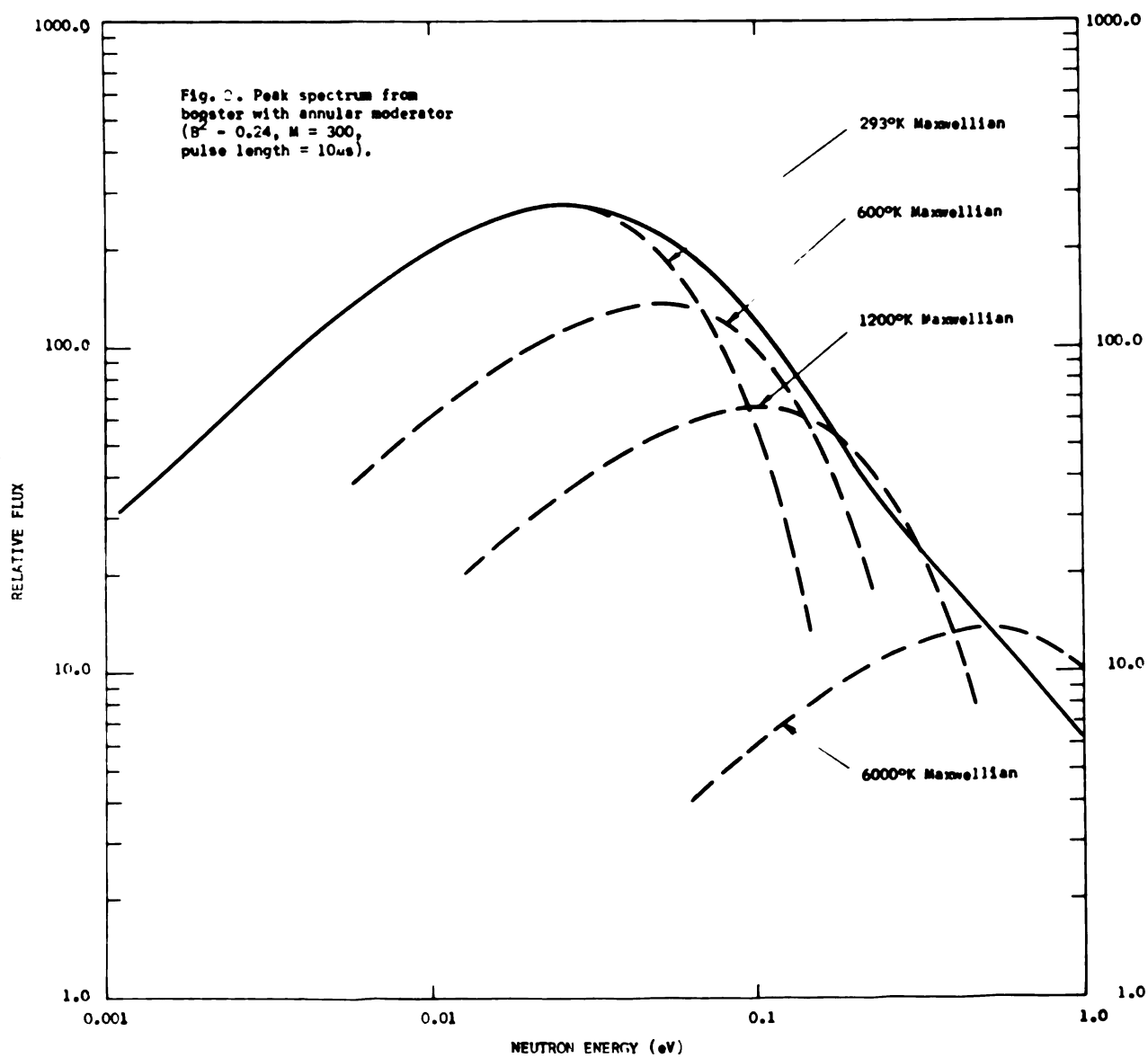


Fig. 2

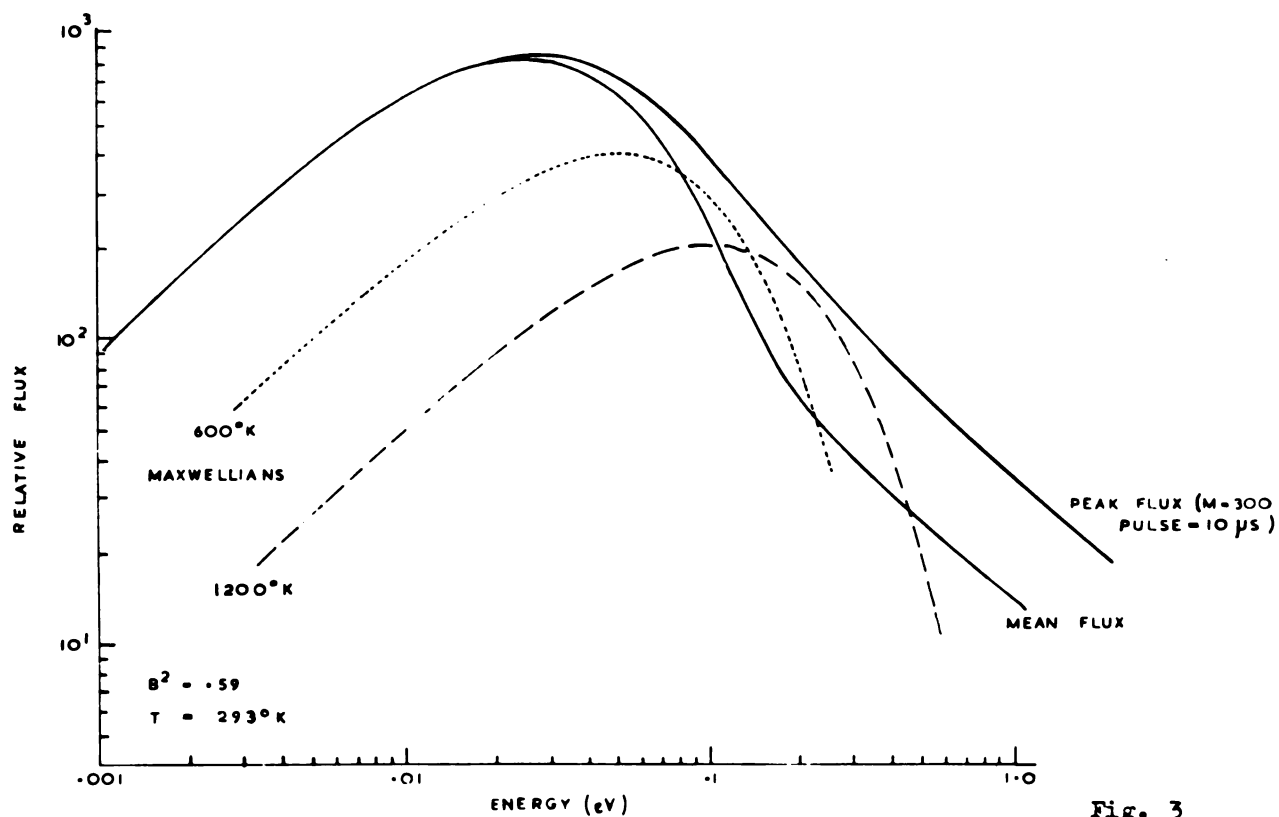
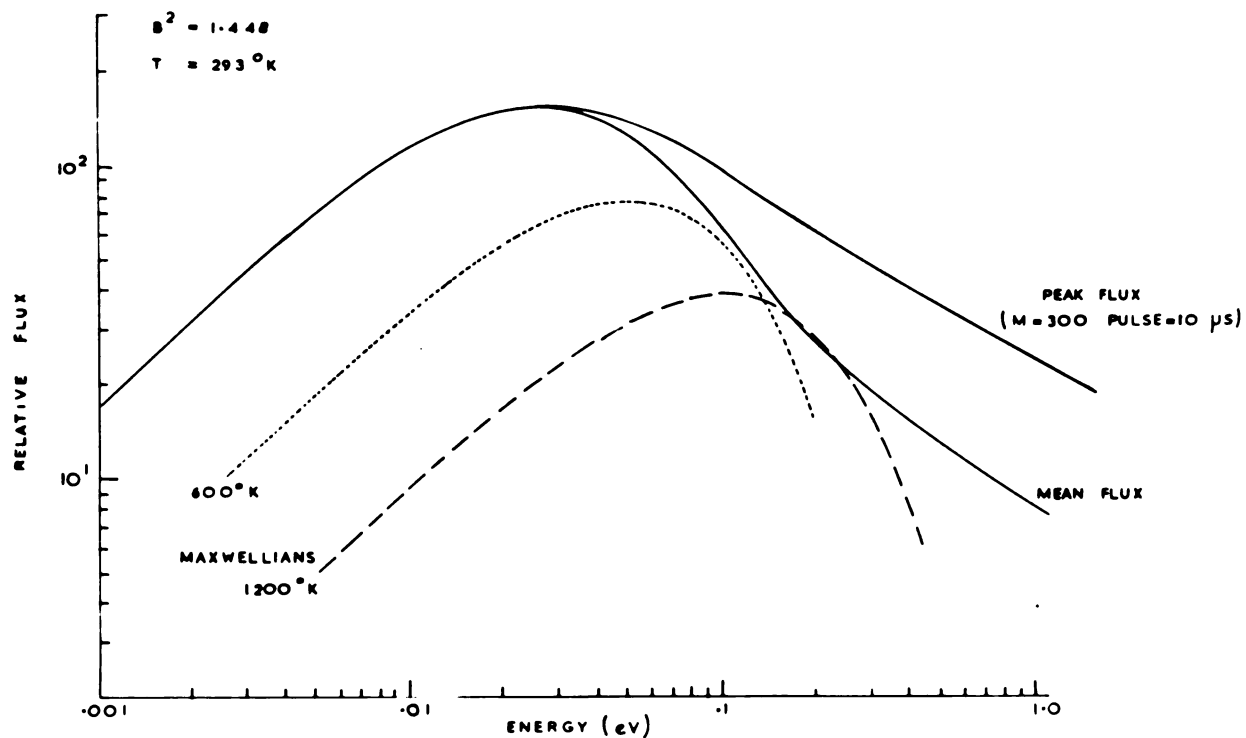


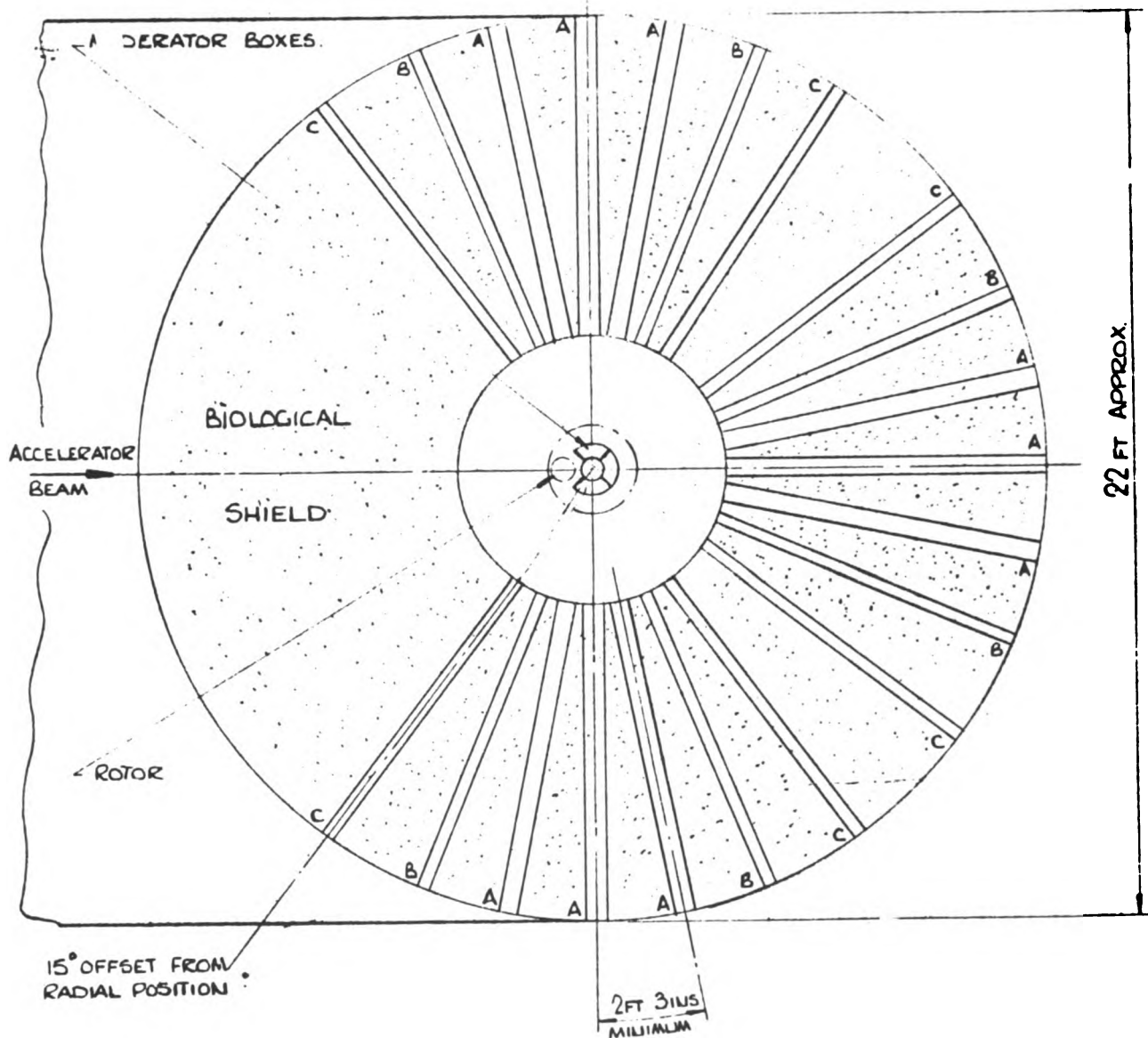
Fig. 3

A.E.R.E. NP-GEN/45. FIG. 3. STATIONARY SPECTRUM FOR SUPER-BOOSTER MODERATOR.



A.E.R.E. NP-GEN/45. FIG. 4. STATIONARY SPECTRUM FOR SUPER-BOOSTER MODERATOR.

Fig. 4



POSSIBLE FLIGHT TUBE SIZES.

TYPE 'A' - 6 IN. DIA - RADIAL - 9 OFF

TYPE 'B' - 4 IN. DIA - RADIAL - 6 OFF

TYPE 'C' - 4 IN. DIA - ^{OFFSET 15°}_{FROM RADIAL} - 6 OFF

TOTAL 21

SEE FIG. 2. FOR ENLARGED VIEW IN CORE AREA

FLIGHT TUBE DISTRIBUTION IN CONCRETE SHIELDING.

Fig. 6

OPERATING CHARACTERISTICS OF ACCELERATOR-BOOSTER PULSED RESEARCH REACTORS*

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SUMMARY: During the past several months, a feasibility study of an accelerator-booster fast repetitively pulsed research reactor has been performed at General Atomic for the AEC.

Desirable operating characteristics for the booster have been determined, together with its dependence upon the associated Linac. Fast neutron burst shapes, power levels, and problems of thermalizing the fast neutrons to obtain short intense bursts of thermal neutrons are discussed. Alternative reactor core, coolant, and reflector materials have been examined for possible use in the booster.

Briefly, the principle of operation is as follows. A rotating reactor component varies the reactivity of a fast reactor periodically between a peak which is below prompt critical and a minimum of a few dollars subcritical. At the high point of the reactivity swing, a linear accelerator produces a burst of neutrons in the reactor. The booster then multiplies this neutron intensity by a factor which exceeds 100. The fast neutrons leak into a hydrogenous moderator which is located outside of the fast core.

The result is an intense burst of thermal neutrons which is available for time-of-flight experiments such as neutron diffraction or inelastic scattering. The reactivity is low between pulses, so that the power level in the reactor between pulses is low, as is the neutron background.

The motivation for this type of research reactor is that large time dependent thermal neutron fluxes can be produced in a reactor which operates at a relatively low average power level. In this way heat removal, shielding, fuel burnup, and other problems associated with high power reactors can be minimized.

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A typical reactor configuration for the booster consists of a high fuel density cylindrical reactor core with a Linac target in the middle. A reflector made of tungsten, nickel alloy, or stainless steel surrounds the core. The moderator is placed as close to the core as possible, but a boron sheet is inserted between the two. This prevents the return of thermal neutrons from the moderator into the core in order to maintain a short neutron generation time. For experiments with low energy neutrons, it is advantageous to poison the moderator in order to reduce the dieaway time.

There are a number of ways to vary the reactivity periodically. Some move fuel or absorbing material into the core with a wheel; others move neutron reflecting material next to the core. A gas coolant can be used for average power levels up to 0.5 MW, preferably CO₂. For higher power levels, it is necessary to use a liquid metal coolant. The best choices are sodium or NaK.

A typical booster can produce peak thermal neutron fluxes in excess of 10^{15} neutrons/sec cm² in pulses of 20 μ sec width at a repetition rate of 100 pulses per second if operated at an average power level of 0.5 MW. Of course, these parameters can be traded off against each other, so that they are optimized for any specific experiment.

1. INTRODUCTION

During the past several months, a feasibility study of an accelerator-booster repetitively pulsed research reactor has been performed at General Atomic for the AEC. The work builds on earlier ideas which have been reported in the past (1, 2). The principle of operation is as follows:

A rotating reactor component varies the reactivity of a fast reactor periodically between a peak which is below prompt critical and a minimum of a few dollars subcritical. At the high point of the reactivity swing, a linear accelerator produces a burst of neutrons in the reactor. The booster then multiplies this neutron intensity by a factor which exceeds 100. The fast neutron burst hits a hydrogenous moderator which is located outside of the fast core.

The result is an intense burst of thermal neutrons which is available for time-of-flight experiments such as neutron diffraction or inelastic scattering. The reactivity is low between pulses, so that the power level in the reactor between pulses is low, as is the neutron background.

The motivation for this type of research reactor is that large thermal neutron fluxes can be produced in a reactor which operates at a relatively low average power level. In this way heat removal, shielding, fuel burnup, and other problems associated with high power reactors can be minimized.

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These concepts are discussed in greater length in the following sections.

2. FAST NEUTRON BURST SHAPE

In order to make predictions concerning the performance characteristics of accelerator boosters, consider a simple model of the neutron kinetic behavior. Define the following symbols:

$n(t)$ = neutron population at time t

S = intensity of Linac source (neutrons/sec); it is assumed that weighting factors which arise from the geometries of the reactor and the Linac target have been included.

ρ_a = reactivity during the Linac pulse

ρ_b = background reactivity

T_A = time during which reactivity is ρ_a . It is assumed that at $t = 0$, the reactivity changes from ρ_b to ρ_a .

τ_f = effective fast neutron burst width.

T_0 = time that Linac is turned on

T_e = time that Linac is turned off

$T_s = T_e - T_0 = \text{Linac on-time.}$

$\beta = \text{delayed neutron fraction}$

$\ell = \text{neutron generation time}$

$$A = \frac{\beta - \rho_a}{\ell}$$

$$B = \frac{\beta - \rho_b}{\ell}$$

$\nu = \text{average number of neutrons produced per fission}$

$T = \text{period} = 1/\text{pulse repetition rate}$

$n_A(t) = \text{neutron population due to accelerator source}$

$n_B(t) = \text{neutron population due to delayed neutrons}$

$\bar{n}, \bar{n}_A, \bar{n}_B = \text{averages of } n, n_A, n_B \text{ over one period}$

$C = \text{precursor concentration}$

We begin with the point kinetic equations

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n + \sum_i \lambda_i C_i + S \quad (1)$$

$$\frac{dC_i}{dt} = -\lambda_i C_i + \frac{\beta_i n}{\ell} \quad (2)$$

Assume that the pulse repetition rate is sufficiently high so that the precursors may be treated as a constant background source. Then

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n + \frac{\beta \bar{n}}{\ell} + S \quad (3)$$

where \bar{n} is the average neutron density over a period.

As the initial condition, let

$$\frac{dn(0-)}{dt} = 0 \quad (4)$$

so that

$$n(0) = \frac{\beta \bar{n}}{\ell B} \quad (5)$$

This condition is consistent with the assumptions that

$$T_e \leq T_A \ll T \quad (5a)$$

and

$$T \gg \frac{1}{B} . \quad (5b)$$

With

$$n(t) = n_A(t) + n_B(t) \quad (6)$$

the resulting solution for $n(t)$ may be obtained from

$$n_A(t) = 0 \quad 0 < t < T_0 \quad (7)$$

$$n_A(t) = \frac{S}{A} (1 - e^{-A(t-T_0)}) \quad T_0 < t < T_e$$

$$n_A(t) = \frac{S}{A} (1 - e^{-AT_s}) e^{-A(t-T_e)} \quad T_e < t < T_A$$

$$n_A(t) = \frac{S}{A} (1 - e^{-AT_s}) e^{-A(T_A-T_e)} e^{-B(t-T_A)} \quad T_A < t < T \quad (8)$$

and

$$n_B(t) = \frac{\bar{n}\beta}{\ell} \left[\frac{1}{A} (1 - e^{-At}) + \frac{e^{-At}}{B} \right] \quad 0 < t < T_A \quad (9)$$

$$n_B(t) = \frac{\bar{n}\beta}{\ell} \left[\frac{1}{B} + \left(\frac{1}{A} - \frac{1}{B} \right) (1 - e^{-AT_A}) e^{-B(t-T_A)} \right] \quad T_A < t < T$$

where

$$\bar{n} = \frac{\frac{S}{AT} \left\{ T_s - \left(\frac{1}{A} - \frac{1}{B} \right) e^{-A(T_A-T_e)} (1 - e^{-AT_s}) \right\}}{1 - \frac{\beta}{\ell T} \left[\frac{T}{B} + T_A \left(\frac{1}{A} - \frac{1}{B} \right) - \left(\frac{1}{A} - \frac{1}{B} \right)^2 (1 - e^{-AT_A}) \right]} . \quad (10)$$

We can define the background ratio by

$$B_R = \frac{\bar{n}_B}{\bar{n}} = \frac{\beta}{\ell T A^2} \left[AT_A + \frac{A^2}{B} (T - T_A) - \left(1 - \frac{A}{B} \right)^2 (1 - e^{-AT_A}) \right] \quad (11a)$$

which is very well approximated by

$$B_R = \frac{\beta}{\beta + |\rho_b|} \quad (11b)$$

and an effective pulse width by

$$\tau_f = \frac{\bar{n}_A T}{n_{A, \max}} = \frac{T_s}{-AT_s} - \left(\frac{1}{A} - \frac{1}{B} \right) e^{-A(T_A - T_e)} \quad (12)$$

A source density ratio, first introduced by Misenta(3), is defined by

$$R_D = \frac{S}{\frac{\beta}{\ell} \bar{n}} = \frac{S \times 3.3 \times 10^{-17}}{\beta \nu P} \quad (13)$$

where P is the average power in MW. By using Eqs. (10) to (13), R_D can be expressed as

$$R_D = \frac{\ell T (1 - B_R) A}{\beta \tau_f (1 - e^{-AT_s})} \quad (14)$$

If we now assume that

$$A(T_A - T_e) \gg 1 \quad (15)$$

we obtain an approximate expression for the pulse width from Eq. (12),

$$\tau_f = \frac{T_s}{-AT_s} \frac{1}{1 - e^{-AT_s}}, \quad (16)$$

which is consistent with the assumption that the reactivity is still at a high point for some time after the Linac has been turned off.

The energy in a pulse is

$$E = PT(1 - B_R). \quad (17)$$

From Eqs. (13) and (14) this becomes

$$E = \frac{3.3 \times 10^{-17} S (1 - e^{-AT_s}) \tau_f}{\nu l A} \quad (18)$$

With the help of Eq. (16), this simplifies to

$$E = \frac{3.3 \times 10^{-17} S T_s}{\nu l A} \quad (19)$$

In Eq. (19), S is the Linac source strength in n/sec, and E is the energy in MW seconds.

If we wish to maximize the energy in a pulse, for a fixed pulse width τ_f , we proceed as follows. From Eqs. (16) and (19) we obtain

$$E = - \frac{3.3 \times 10^{-17} S T_s^2}{\nu l \ln \left(1 - \frac{T_s}{\tau_f}\right)} \quad (20)$$

Differentiating with respect to T_s , and setting the derivative equal to zero leads to the requirement for the maximum energy in the pulse:

$$\frac{X}{1 - X} = 2 \ln \frac{1}{1 - X} \quad (21)$$

where

$$X \equiv \frac{T_s}{\tau_f} \quad (22)$$

This is satisfied when

$$T_s = .715 \tau_f \quad (23)$$

From Eq. (16), it also follows that for this optimum situation

$$A \tau_f = 1.755, \quad (24)$$

and

$$A T_s = 1.255 \quad (25)$$

Insertion of these relations into Eq. (19) gives:

$$E = \frac{1.35 \times 10^{-17} S \tau_f^2}{\nu l} \quad (26)$$

Equation (26) is an expression which gives the maximum energy which can be generated in a pulse, for a given fast burst width. No other combination of Linac-on-time and reactivity can give a higher energy without increasing the fast burst width. For long fast reactor burst widths this optimum condition cannot be met because the maximum Linac-on-time in a real Linac is less than the required value indicated in Eq. (23). For optimum operation, the reactivity during the pulse should be set to:

$$\rho_a = \beta - 1.755 \frac{l}{\tau_f} \quad (27)$$

The power in the pulse, averaged over the period, T , is just

$$P_A = \frac{E}{T} = \frac{1.35 \times 10^{-17} S \tau_f^2}{\nu l T} \quad (28)$$

The total power, when the reactor is operated at the optimum condition, is

$$P = \frac{\beta + |\rho_b|}{|\rho_b|} P_A \quad (29)$$

We now compare two reactors, one fueled with Pu-239, the other with U-235. In each case, we assume that $l = 2 \times 10^{-8}$ sec, $T = 0.01$ sec, and $S = 10^{17}$ n/sec. For U-235, we take $\nu = 2.47$ and $\beta = .0065$. For Pu-239, we have $\nu = 2.9$ and $\beta = .002$. A reactivity changing wheel is assumed to have a total reactivity swing of 0.02. The comparison is shown in Table I. It can be seen that the principal advantage of the Pu-239 over the U-235 is the lower background ratio. To produce power pulses in the two reactors having approximately the same intensity and pulse width, a larger total power must be produced in the U-235 system. The necessity of raising the reactivity to a higher level in the U-235 system than in the Pu-239 system, for the same pulse characteristics, is not a serious disadvantage if a wheel is available. It is interesting to note that for the Pu-239 system, it is only necessary to swing the wheel above delayed critical for the 20- μ sec fast burst width. This, of course, assumes that Linac-on-times indicated in the tables are realizable.

If a booster were operated without a wheel, these considerations change only slightly. Equations (1) to (8) still apply, but this time $\rho_a = \rho_b$. It is not likely that we would want the background ratio to exceed 0.5. Thus, for the Pu-239 system the maximum reactivity will be limited to -0.002, while for the U-235 system it will be limited to -0.0065. Table II gives the characteristics of the Pu-239 and U-235 systems, each with a constant reactivity set to -\$.1.00.

Table I
PERFORMANCE CHARACTERISTICS OF
Pu-239 AND U-235 BOOSTERS, OPERATING AT OPTIMUM CONDITIONS

Burst Width τ_f	Power in Pulses P_A (MW)	Average Power P (MW)	Linac-on-Time T_s (μ sec)	Reactivity During Pulse ρ_a	Background Reactivity ρ_b
<u>Pu-239</u>					
5	.058	.0626	3.57	-.005	-.025
10	.2322	.254	7.15	-.00151	-.02151
15	.522	.573	10.73	-.00034	-.02034
20	.929	1.02	14.3	.00025	-.01975
<u>U-235</u>					
5	.0681	.0897	3.57	-.00052	-.0205
10	.273	.3768	7.15	.00299	-.017
15	.613	.864	70.73	.00416	-.0158
20	1.09	1.54	14.3	.00425	-.0157

Table II
OPTIMUM PERFORMANCE CHARACTERISTICS OF U-235
AND Pu-239 BOOSTERS OF CONSTANT REACTIVITY
EQUAL TO -\$.1.00

Fuel	τ_f (μ sec)	Power in Pulses P_A (MW)	Average Power P (MW)	Linac-on-Time T_s (μ sec)	Reactivity During Pulse ρ_a
Pu-239	8.775	.18	.36	6.27	-.002
U-235	2.7	.02	.04	1.93	-.0065
U-235	5.2	.052	.104	5	-.0065

The optimum Linac-on-time for the U-235 system, consistent with the above discussion, is only 1.93 μ sec. This leads to an average power of only 40 kw, which is much too small. If instead of this "optimum operation", we leave the Linac on for 5 μ sec, which is about the maximum of any Linac now in operation, we get an average power operation of 0.104 MW of which half goes into the pulse. For higher powers, we need a larger Linac or a wheel.

From Table I it can be seen that for optimum operation with a Linac having a source strength of 10^{17} n/sec and a repetition rate of 100 pulses per second, the power does not exceed 2 MW even for a relatively long pulse of 20 μ sec. As Linacs with more source intensity become available, larger booster power would be desirable. In the design of these larger boosters, much effort will be needed to attain the larger average power without paying a heavy price in neutron lifetime. This will require higher and higher power densities, together with materials capable of taking the higher resulting temperatures. For the time being, however, the ideal average powers are about 0.5 - 2 MW. It is certainly feasible, with present technology, to build a booster of that power capacity having a neutron lifetime of .01 to .02 μ sec.

This discussion has been limited to a conceptional booster having a fixed repetition rate of 100 pulses per second. A booster can easily be designed to have a variable frequency, and this would lead to experimental advantages. Some of these are discussed in the following section.

3. ADVANTAGES OF SHORT FAST NEUTRON BURST SHAPE

In the previous section we have concerned ourselves primarily with the question of how to optimize the energy in a pulse for a fixed pulse width. Here we discuss the criteria by which the desired value of the pulse width is to be chosen.

We begin with some experimental considerations. Generally the count rate in a time-of-flight experiment may be expressed by the proportionality relation

$$N = K_1 \frac{\phi_{\max} \tau}{L^2 T} \quad (30)$$

where τ is the pulse width, and L is the source-detector distance. For experiments with low energy neutrons, it is usual to assume, within limits, that the pulse repetition rate can be adjusted and that it is inversely proportional to the source-detector distance. Accordingly,

$$L = K_2 T \quad (31)$$

The value of K_2 depends upon additional information such as, for instance, whether or not a Be filter is part of the experimental setup. Combining Eqs. (30) and (31) gives

$$N = K_1 K_2 \frac{\varphi_{\max} \tau}{L^3} \quad (32)$$

The resolution is inversely proportional to the distance:

$$R = \frac{\Delta E}{E} = \frac{2\tau v}{L} \quad (33)$$

where v is the velocity of neutrons used in the experiment.

From Eqs. (31) and (33), we define the duty cycle

$$K_3 = \frac{\tau}{T} = \frac{K_2 R}{2v} \quad (34)$$

Inserting Eq. (33) into (32) gives

$$N = \frac{K_1 K_2 \varphi_{\max}}{8v} \frac{R^3}{\tau^2} \quad (35)$$

Since it is desired to perform the experiment at some fixed resolution, the counting rate is proportional to φ_{\max}/τ^2 , which we shall call the figure of merit, F . Thus

$$F = \frac{\varphi_{\max}}{\tau^2} \quad (36)$$

For any experiment it is desirable to maximize F . φ_{\max} can be estimated from

$$\varphi_{\max} = \frac{\bar{\varphi} T}{\tau} = \frac{\bar{\varphi}}{K_3} \quad (37)$$

where $\bar{\varphi}$ is the average flux over a period, but not including the background flux. Combining the last two expressions gives

$$F \approx \frac{\bar{\varphi}}{\tau} \quad (38)$$

To relate the total pulse width, τ , to the fast neutron burst width, τ_f , and the pulse width in the moderator due to a delta function source, τ_m , we use the following argument.

Suppose that $G(t)$ is the thermal pulse shape due to a delta function fast source. Furthermore, let $P(t)$ be the actual fast source shape. Then the actual thermal pulse shape is

$$\Phi(t) = \int_0^t G(t-\tau) P(\tau) d\tau \quad (39)$$

Let $\bar{\Phi}(s)$ be the Laplace transform of $\Phi(t)$, with a similar notation for G and P . Then

$$\bar{\Phi}(s) = \int_0^{\infty} \Phi(t) e^{-st} dt \quad (40)$$

and

$$\bar{\Phi}(s) = \bar{G}(s) \bar{P}(s) \quad (41)$$

The moments of $\Phi(t)$, Φ_n , are defined by

$$\Phi_n = \frac{1}{n!} \int_0^{\infty} t^n \Phi(t) dt \quad (42)$$

and it is easily seen that this is equivalent to

$$\Phi_n = \frac{(-1)^n}{n!} \left. \frac{\partial^n \bar{\Phi}(s)}{\partial s^n} \right|_{s=0} \quad (43)$$

The moments of $G(t)$ and $P(t)$, are defined in a similar fashion, so, proceeding formally,

$$\Phi_0 = G_0 P_0 = \bar{G}(0) \bar{P}(0), \quad (44)$$

$$\Phi_1 = \left. \frac{-\partial \Phi(s)}{\partial s} \right|_{s=0} = - \left[G_0 \left. \frac{\partial P(s)}{\partial s} \right|_{s=0} + P_0 \left. \frac{\partial G(s)}{\partial s} \right|_{s=0} \right]$$

$$\Phi_1 = G_0 P_1 + P_0 G_1 \quad (45)$$

$$\Phi_2 = \frac{1}{2} \left[G_0 \left. \frac{\partial^2 P}{\partial s^2} \right|_{s=0} + 2 \left. \frac{\partial G}{\partial s} \right|_{s=0} \left. \frac{\partial P}{\partial s} \right|_{s=0} + P_0 \left. \frac{\partial^2 G}{\partial s^2} \right|_{s=0} \right] \quad (46)$$

$$\Phi_2 = G_0 P_2 + G_1 P_1 + P_0 G_2$$

The mean of the pulse shape $\Phi(t)$, which physically is the propagation time, is

$$\bar{t}_\Phi = \frac{\Phi_1}{\Phi_0} = \frac{G_0 P_1 + P_0 G_1}{G_0 P_0} = \frac{P_1}{P_0} + \frac{G_1}{G_0} = \bar{t}_P + \bar{t}_G \quad (47)$$

Similarly, \bar{t}_Φ^2 may be computed from

$$\bar{t}_\Phi^2 = \frac{2\Phi_2}{\Phi_0} = \frac{2G_0 P_2 + 2G_1 P_1 + 2P_0 G_2}{G_0 P_0} = \frac{2P_2}{P_0} + \frac{2G_1 P_1}{G_0 P_0} + \frac{2G_2}{G_0} \quad (48)$$

or

$$\bar{t}_\Phi^2 = \bar{t}_P^2 + 2\bar{t}_G \bar{t}_P + \bar{t}_G^2$$

The variance σ_Φ^2 is

$$\sigma_\Phi^2 = \bar{t}_\Phi^2 - \bar{t}_\Phi^2 = (\bar{t}_P^2 - \bar{t}_P^2) + (\bar{t}_G^2 - \bar{t}_G^2) = \sigma_P^2 + \sigma_G^2 \quad (49)$$

The standard deviation, σ_Φ , can now be computed from

$$\sigma_\Phi = \sqrt{\sigma_P^2 + \sigma_G^2} \quad (50)$$

For any reasonably shaped pulse, it is qualitatively correct to assume that

$$\frac{\tau_f}{\sigma_P} = \frac{\tau_m}{\sigma_G} = \frac{\tau}{\sigma_\Phi} \quad (51)$$

so that

$$\tau = \sqrt{\tau_f^2 + \tau_m^2} \quad (52)$$

In this way, we have derived a simple relation among the pulse width in the moderator, τ , the fast neutron burst width, τ_f , and the pulse width of neutrons in the moderator due to a delta function source, τ_m . If desired, the value of τ_m can be changed by changing the size of the moderator, or by poisoning it. τ_f is adjustable by the means discussed in the previous section.

From Eqs. (26), (34), (38), and (52), we can express the figure of merit by

$$F = \text{const} \frac{\tau_f^2 Z}{(\tau_f^2 + \tau_m^2)^{3/2}} \quad (53)$$

The parameter Z has been introduced to account for the loss of average thermal flux when poison is added to the moderator. It is defined as the ratio of average flux which is present in a poisoned moderator to that which would be present in the unpoisoned moderator. Both Z and τ_m depend on poison concentration, of course, but unfortunately, that dependence is not expressible in terms of simple analytic formulae. However, both Z and τ_m tend to vary in the same way with varying poison concentration. This can be inferred from Eq. (44) and Figs. 1 and 2, for instance. The method of calculation which was used to arrive at Figs. 1 and 2 is described in the following section. Since Z/τ_m tends to remain constant, we may write.

$$F = \text{const} \frac{\tau_f^2 \tau_m}{(\tau_f^2 + \tau_m^2)^{3/2}} \quad (54)$$

If we had not derived Eq. (26) from Eq. (20) by first adjusting T_s , we could have written the more general expression

$$F = \text{const} \frac{T_s^2 \tau_m}{\ln\left(1 - \frac{T_s}{\tau_f}\right) (\tau_f^2 + \tau_m^2)^{3/2}} \quad (55)$$

Note that setting $\frac{\partial F}{\partial T_s} = 0$ gives us Eq. (54) again, as expected. It is desired to maximize F by varying the parameters T_s , τ_m , τ_f . From Eqs. (31), (33), and (34), it can be seen that these are related to the resolution, repetition rate, and length of flight path. Thus, F is to be maximized, but certain constraints on T_s , τ_m and τ_f must be imposed. First, the Linac-on-time has some upper limit

$$0 < T_s \leq T_{s, \max} \quad (56)$$

The flight path length L is restricted

$$L_{\min} \leq L \leq L_{\max} \quad (57)$$

where L_{\min} is determined by shielding requirements and other considerations. L_{\max} is limited possibly by the cost and inconvenience of a long flight path. The frequency has an upper limit which is fixed by the maximum repetition rate of the accelerator, or the mechanical limitations imposed on a rotating reactivity-changing wheel,

$$T \geq T_{\min} \quad (58)$$

τ_m has an upper limit at each energy which corresponds to the value in the unpoisoned moderator,

$$\tau_m < \tau_{m0} \quad (59)$$

The average power in the pulses must be less than some maximum value, which is imposed by heat removal limitations,

$$P_A \leq P_{A, \max} \quad (60)$$

If a wheel is used, we must have $\rho_a < \beta$. If there is no wheel, $\rho_a = \rho_b \leq -\beta$. The limit on ρ_b has been arbitrarily chosen so that the background power fraction does not exceed 0.5. It can be argued that this limit is too high, of course.

It is desired to find values of the independent parameters which maximize F and which are consistent with the constraints which are imposed by Eqs. (56) to (60), as well as those imposed upon the reactivity.

If two or more experiments, having different requirements, use the facility simultaneously, the problem becomes more complicated. Then there is a figure of merit, having the form of Eq. (55), for each experiment. Some parameters such as R , L , and τ_m can be adjusted separately for each experimenter. But others, such as P_A , T_s , T , ρ_a , and τ_f are common to all the experiments. This introduces more independent variables, but still more constraints. As more experimenters are added, the flexibility thus decreases. It may then become impossible to find a true maximum for each F ; that is, the operating conditions which optimize one experiment are in conflict with those which optimize another. Additional weighting factors, which may involve the relative scientific value of the experiments, could well become important criteria at this stage. A weighted sum of the individual F 's could then be optimized.

Before proceeding, it is perhaps best to review what the adjustable parameters are, and to put together those equations which relate them. The parameters are T_s , τ_m , τ_f , τ , T , L , P_A , P , ρ_a , ρ_b , E , R , and A . These are related by the definition of A , Eqs. (16), (19), (29), (31), (33), and (52), and the definition $P = E/T$. Furthermore, $\rho_a - \rho_b = \Delta\rho$ is fixed. Thus, there are thirteen parameters and nine equations which relate them. In addition, the resolution is arbitrarily fixed by the experimenter. This leaves three degrees of freedom with which to maximize F . In addition to the nine equations, the parameters are further restricted by the constraints imposed by Eqs. (56) to (60).

Equation (55) relates F to the three parameters T_s , τ_m , and τ_f . It is probably more natural to use T_s , P_A , and T as independent parameters. This is so because the constraints on these are well defined, and they form a simple volume in (T_s, P_A, T) space. In addition, their physical interpretation is clear. The additional constraints can eliminate portions of (T_s, P_A, T) space as possible regions of operation. These portions will differ, in general, for different experiments.

Combining Eqs. (16), (19), (34), (52), and (55) enables the rewriting of Eq. (55) as

$$F = \text{const} \frac{P_A \tau_m}{T^2} \quad (61)$$

where

$$\tau_m = \sqrt{K_3^2 T^2 - \frac{T_s^2}{\left[1 - \exp\left(-\frac{K_4 T_s^2}{TP_A}\right)\right]^2}} \quad (62)$$

and

$$K_4 = \frac{3.3 \times 10^{-17} S}{\nu \ell} \quad (63)$$

Equation (61) has a particularly simple interpretation if τ_m is not adjustable. It says that the best way to operate is to run at as high power as possible, and as high a repetition rate as is possible, subject to the limitations imposed by the constraints of Eqs. (56) to (60).

If τ_m is allowed to vary we have, of course,

$$F = \text{const} \frac{P_A}{T^2} \sqrt{K_3^2 T^2 - \frac{T_s^2}{\left[1 - \exp\left(-\frac{K_4 T_s^2}{TP_A}\right)\right]^2}} \quad (64)$$

Now, $\frac{\partial F}{\partial T_s} = 0$ when

$$e^x = 1 + 2x; x = 1.255 \quad (65)$$

where

$$x = \frac{K_4 T_s^2}{TP_A} \quad (66)$$

Thus,

$$T_s = 1.12 \sqrt{\frac{TP_A}{K_4}} \quad (67)$$

Eq. (67), of course, is equivalent to Eq. (23).

Inserting Eq. (67) into (64) yields

$$F = \text{const} \frac{P_A}{T^2} \sqrt{K_3^2 T^2 - \frac{2.45 T P_A}{K_4}} \quad (68)$$

$$\frac{\partial F}{\partial P_A} = 0 \text{ when}$$

$$P_A = \frac{2 K_3^2 T K_4 (1 - e^{-x})^2}{3 x} ;$$

that is, when

$$P_A = 0.272 K_3^2 K_4 T \quad (69)$$

Then, by inserting Eq. (69) into (68),

$$F = \text{const} \quad (70)$$

This leads to the interesting conclusion that there is no single point within the enclosed volume of (T_s, P_A, T) space at which F is optimized. Instead, any point on the line defined by Eq. (69) and

$$T_s = 0.585 K_3 T, \quad (71)$$

which is obtained from Eqs. (67) and (69), yields the same maximum figure of merit.

If any one of the three parameters is fixed by a particular experimental circumstance or some other criterion, the optimum operating condition is completely determined - unless one of the constraints on the other variables is violated. It is also of interest to know the optimum operating conditions if two of

the parameters are fixed. Three examples are

1. P_A , T fixed. T_s is variable.

Eq. (67) gives the optimum value of T_s .

2. P_A , T_s fixed. T is variable.

$$T = \frac{K_4 T_s^2}{x P_A} \quad (72)$$

where x is the root of

$$y^2 (1 - e^{-x})^3 + x^3 e^{-x} = 2x^2 (1 - e^{-x}) \quad (73)$$

and

$$y \equiv \frac{K_3 K_4 T_s}{P_A} \quad (74)$$

3. T_s , T fixed. P_A is variable.

$$P_A = \frac{K_4 T_s^2}{x T} \quad (75)$$

where x is the root of

$$\frac{x e^{-x}}{1 - e^{-x}} = y^2 (1 - e^{-x})^2 - 1 \quad (76)$$

and

$$y \equiv \frac{K_3 T}{T_s} \quad (77)$$

The optimum figure of merit obtainable by operating according to these restricted conditions will not be as high, of course, as that which is obtainable by operating on the line defined by Eqs. (69) and (71).

It is easily shown that, consistent with Eqs. (69) and (71), the pulse widths are related by:

$$\tau_m = 0.99 T_s = .707 \tau_f \quad (78)$$

Equation (78) indicates that to obtain the maximum figure of merit, the important pulse widths should be approximately equal. This conclusion is consistent with what one would have guessed intuitively. Of course, from Eq. (34), we know that the period is also proportional to these pulse widths.

Insofar as obtaining the maximum figure of merit is concerned, nothing is said of the absolute values of the time constants, only of their ratio. A strong inducement for operating at short time constants is seen from Eq. (69), where we see that the average power goes in proportion to the time constants. Thus, the average power is lower, for the same figure of merit, at short pulse operation than at long pulse operation. This is a decided advantage in favor of operating with short flight paths, high repetition rates and, if necessary to reduce τ_m , with poisoned moderators.

A mechanically pulsed reactor has a fast neutron burst width which exceeds, by far, the width obtainable with an accelerator booster. From the above discussion, it appears that a principal advantage of the latter could be the capability of obtaining the same figure of merit as the former, at lower average powers. Equivalently, the booster can have a higher figure of merit than the mechanically pulsed reactor at the same power. This is being looked at further.

It is useful to put together, in one place, the values of the important parameters which are obtained when operating on the line of optimum figure of merit, defined by Eqs. (69) and (71). Since there is one free parameter, the results have arbitrarily been tabulated in terms of T_s . The results are:

$$\begin{aligned}
 f &\equiv \frac{1}{T} = \frac{0.292}{T_s} \frac{K_2 R}{v} \\
 P_A &= 0.232 \left(\frac{K_2 R}{v} \right) \left(\frac{3.3 \times 10^{-17} \text{ S}}{\nu \ell} \right) T_s \\
 \tau_m &= 0.99 T_s \\
 \tau_f &= 1.4 T_s \\
 \tau &= 1.71 T_s \\
 L &= 3.42 \frac{v}{R} T_s
 \end{aligned} \tag{79}$$

$$\rho_a = \beta - 1.255 \frac{\ell}{T_s}$$

$$P = 0.232 \left(\frac{K_2 R}{v} \right) \left(\frac{3.3 \times 10^{-17} \text{ S}}{\nu \ell} \right) \frac{1.255 \frac{\ell}{T_s} + \Delta \rho}{\left(-\beta + 1.255 \frac{\ell}{T_s} + \Delta \rho \right)} T_s$$

$\Delta \rho$ is the reactivity worth of the reactivity wheel.

The optimum source density ratio, defined by Eq. (13), can also be evaluated easily. It is an indicator of how much of the power is to be obtained from the accelerator, relative to that obtained from the reactor. We obtain

$$R_D = 4.3 \frac{\ell}{\beta} \frac{v}{K_2 R} \frac{1.255 \frac{\ell}{T_s} + \Delta \rho - \beta}{1.255 \frac{\ell}{T_s} + \Delta \rho} \frac{1}{T_s} \quad (80)$$

Here we see that, as the neutron energy of interest increases, we want more accelerator and less reactor. The fact that R_D decreases as $\Delta \rho$ decreases is a consequence of ignoring background ratio considerations in our figure of merit.

In the above equations the velocity of the neutrons of interest enters as the ratio $\frac{v}{K_2 R}$ except for the expression for L , in which case it is just $\frac{v}{R}$. K_2 , the ratio of flight path length to period, can be varied from one type of experiment to another. Typically, it is about 2000 m/sec, but its value can only be determined from additional information such as whether or not a Be filter, or a slow chopper, etc., is part of the experimental setup.

As an example of the use of Eq. (79), consider an experiment in which neutrons of $v = 2200$ n/sec are of primary interest. We also assume a plutonium system and the values: $\beta = .002$, $R = 0.01$, $S = 10^{17}$ n/sec, $\nu = 2.9$, $\ell = 2.0 \times 10^{-8}$ sec, $K_2 = 2000$ m/sec, $T_s = 5 \mu\text{sec}$. We conclude that for optimum operation, $f = 531$ pps, $P_A = 0.6$ MW, $\tau_m = 4.95 \mu\text{sec}$ (which implies poisoning a water moderator of $B^2 = 0.1755$ by about 15 times the natural absorption), $\tau_f = 7 \mu\text{sec}$, $\tau = 8.62 \mu\text{sec}$, $L = 3.76$ meters, $\rho_a = -.003 = -1.5\beta$. The optimum frequency is rather high but, for the sake of argument, we assume that our Linac is capable of it. We can make the following observations. For this experiment, the high frequency probably precludes the use of a wheel but the background ratio is tolerable without one -

the total power being 1.0 MW. A smaller frequency would necessitate a longer T_s but, at $T_s = 5 \mu\text{sec}$, we assume that we are pushing the limit. If we had used U^{235} instead of Pu^{239} , the larger value of β would force ρ_a to be positive, and a wheel would be necessary. This would probably limit our maximum frequency to well below 500 pps and, since we are already at the limit of Linac-on-time, it is impossible to work along the line defined by Eqs. (69) and (71). We must accept a lower figure of merit. So, we fix $T_s = 5 \mu\text{sec}$, and $f = 100 \text{ pps}$, and compute the optimum power from Eq. (75). The result is 1.02 MW, which we assume to be less than the maximum available power, allowing for background power. Using the same data as above, except $\nu = 2.47$ and $\beta = .0065$, we get $L = 20 \text{ meters}$, $\tau_m = 30.5 \mu\text{sec}$, $\tau_f = 33 \mu\text{sec}$, $\tau = 45 \mu\text{sec}$, $\rho_a = .0059$. The total power will depend on the reactivity worth of the wheel, which has not been specified. These are attainable operating conditions. As can be seen most easily from Eq. (61), the Pu^{239} system without the wheel has a figure of merit which is 2.75 times that of the U^{235} system with the wheel. Furthermore, the Pu system attains the higher figure of merit with a lower average power level. Of course, no sweeping generalizations can be made on the basis of this one example.

4. PULSE SHAPE IN THE MODERATOR

The shape of the pulse of neutrons in the moderator is dependent upon a number of factors. These include: size and shape of the moderator, moderating materials used, quantity and geometrical distribution of additional absorbing materials, energy variation of absorbing cross section, as well as the intensity, time-duration, angular distribution, and energy spectrum of the source of neutrons. From the above it is clear that due to the large number of parameters involved, the design of a moderator capable of producing a specific neutron pulse of a particular energy at a fixed position is an involved engineering problem. The fact that, for any fixed set of parameters, the neutron pulse shape is obtained from a solution of the time, energy, angle, and space-dependent Boltzmann equation is a further confirmation of the complexity of the problem.

It is best to begin the study of the pulse shape in the moderator by making a number of simplifying assumptions. In this way, a tractable problem results, from which we may obtain a feel for the important parameters. Toward this end, consider a homogeneous moderator. Assume that diffusion theory may be used

for the analysis and, therefore, that the geometry and size of the moderator may be characterized simply by the buckling.

The equation we wish to solve is

$$\frac{1}{v} \frac{\partial \Phi(v, t)}{\partial t} = - (DB^2 + \Sigma_T) \Phi(v, t) + \int_0^\infty \Sigma(v' \rightarrow v) \Phi(v', t) dv' + Q(v, t) \quad (81)$$

where the notation is in common use. The solution will be obtained by the moments method. Accordingly, multiply Eq. (81) by $t^n/n!$, integrate over all times and obtain

$$(DB^2 + \Sigma_T) \Phi_n(v) = \int_0^\infty \Sigma(v' \rightarrow v) \Phi_n(v') dv' + \frac{\Phi_{n-1}(v)}{v} + Q_n(v) \quad (82)$$

where the moments of the flux and source are defined by

$$\Phi_n(v) = \frac{1}{n!} \int_0^\infty \Phi(v, t) t^n dt \quad (83)$$

and

$$Q_n(v) = \frac{1}{n!} \int_0^\infty Q_n(v, t) t^n dt \quad (84)$$

In arriving at Eq. (82) it has been assumed that there was no flux present prior to the insertion of the source. Equation (82) is a set of equations, each one having the form of a steady-state equation, by which each moment is obtained recursively from the previous one. The zeroth moment, in particular, which is the total area under the pulse, is the solution of the usual steady-state equation whose steady-state source is our time-integrated source. Since we are primarily interested in thermal neutrons, we are interested in the solution, for $v < v_c$, of the equation

$$(DB^2 + \Sigma_t) \Phi_n(v) = \int_0^{v_c} \Sigma(v' \rightarrow v) \Phi_n(v') dv' + \frac{\Phi_{n-1}(v)}{v} + S_n(v) \quad (85)$$

The thermal source $S_n(v)$ is not actually an external source; the external source $Q_n(v)$ is taken as zero at low velocities. Instead $S_n(v)$ is

$$S_n(v) = \int_{v_c}^{\infty} \Sigma(v' \rightarrow v) \Phi_n(v') dv' \quad (86)$$

where $\Phi_n(v)$ is the solution of

$$(DB^2 + \Sigma_T) \Phi_n(v) = \int_{v_c}^{\infty} \Sigma(v' \rightarrow v) \Phi_n(v') dv' + \frac{\Phi_{n-1}(v)}{v} + Q_n(v). \quad (87)$$

For velocities above v_c , we have assumed that there is no upscattering. In many cases, it is a good assumption to neglect the slowing down time to v_c in comparison to the propagation times below v_c . We could formally do this by taking $S_n(v)$ to be the moment of a delta function in time. Thus we would have

$$S_o(v) = \int_{v_c}^{\infty} \Sigma(v' \rightarrow v) \Phi_o(v') dv' \quad (88)$$

and all other $S_n(v) = 0$. It is not clear, however, that it will always be possible to neglect the slowing down time. For this reason, we now obtain approximate expressions for the moments of the slowing down source.

Instead of solving Eq. (87) directly, we now specialize to a mixture of hydrogen mixed with an absorber whose cross section is $\frac{c}{v}$. We then obtain the moments $\Phi_n(v)$ directly from Eq. (83) where $\phi(v, t)$ is taken from the work of Koppel.⁽⁴⁾ Thus

$$\Phi(v, t) = K v^2 t^{2-\lambda} e^{-(c+av)t} \quad (89)$$

where

$$\begin{aligned} K &= \frac{N}{\Gamma(2-\lambda)} a^{2-\lambda} v_o^{-\lambda}, \\ \lambda &= \frac{2DB^2}{a}, \\ a &= \Sigma_s + DB^2, \end{aligned} \quad (90)$$

Γ denotes the gamma function, and N is the number of neutrons injected with velocity v_0 at time $t = 0$.

From Eqs. (83) and (89) we get

$$\Phi_n(v) = \frac{Kv^2 \Gamma(n+3-\lambda)}{n!(c+av)^{n+3-\lambda}} \quad (91)$$

This leads to the recursion relation

$$\Phi_n(v) = \Phi_{n-1}(v) \frac{n+2-\lambda}{n(c+av)} \quad (92)$$

and

$$\Phi_0(v) = \frac{2 \sum_s N}{va^2} \left(\frac{v}{v_0} \right)^\lambda \frac{1}{\left(1 + \frac{c}{av}\right)^{3-\lambda}} \quad (93)$$

or the flux per unit energy may be written as

$$\Phi_0(E) = \frac{\sum_s N}{Ea^2} \left(\frac{E}{E_0} \right)^{\lambda/2} \left(1 + \frac{c}{a} \sqrt{\frac{m}{2E}} \right)^{-(3-\lambda)} \quad (93a)$$

Equation (93) shows clearly how the epithermal flux is decreased by leakage and capture. From Eq. (92) we can get the mean propagation time, and the variance of the fast flux pulse. They are

$$\bar{t} = \frac{\Phi_1}{\Phi_0} = \frac{3-\lambda}{c+av} \quad (94)$$

and

$$\sigma^2 = \frac{2\Phi_2}{\Phi_0} = \frac{3-\lambda}{(c+av)^2} \quad (95)$$

We can now compute the moments of the thermal source from Eqs. (86), (91) and the hydrogen scattering kernel for a free gas,

$$\sum_s (v' \rightarrow v) = \frac{2 \sum_s}{v'} \left(\frac{v}{v'} \right) \operatorname{erf} \left(\sqrt{\frac{mv^2}{2kT}} \right) \quad (96)$$

The result is

$$S_n(v) = \frac{2 \Sigma_s v K \Gamma(n+3-\lambda)}{n! a(n+2-\lambda) (c + av_c)^{n+2-\lambda}} \operatorname{erf} \left(\sqrt{\frac{mv^2}{2kT}} \right) \quad (97)$$

This leads to the recursion relation,

$$S_n(v) = S_{n-1}(v) \frac{(n+1-\lambda)}{n(c+av_c)} \quad (98)$$

and

$$S_o(v) = \frac{2vN}{v_c} \left(1 + \frac{c}{av_c} \right)^{-(2-\lambda)} \left(\frac{v}{v_o} \right)^\lambda \frac{\Sigma_s}{a} \operatorname{erf} \left(\sqrt{\frac{mv^2}{2kT}} \right) \quad (99)$$

In units of energy this becomes

$$S_o(E) = \frac{N}{E_c} \left(\frac{E_c}{E_o} \right)^{\lambda/2} \frac{1}{\left[1 + \frac{c}{a} \sqrt{\frac{m}{2E_c}} \right]^{2-\lambda}} \frac{\Sigma_s}{a} \operatorname{erf} \left(\sqrt{\frac{E}{kT}} \right) \quad (100)$$

With no leakage or capture, this becomes

$$S_o(E) = \frac{N}{E_c} \operatorname{erf} \left(\sqrt{\frac{E}{kT}} \right) \quad (101)$$

Hence a source computed from Eq. (101), as in GATHER-II⁽⁵⁾ for instance, should be multiplied by the factor

$$P = \left(\frac{E_c}{E_o} \right)^{\lambda/2} \frac{1}{\left[1 + \frac{c}{a} \sqrt{\frac{m}{2E_c}} \right]^{2-\lambda}} \frac{\Sigma_s}{a} \quad (102)$$

if we wish to take into account the loss due to leakage and absorption in the epi-thermal range. Equation (102) is useful for computing absolute intensities of thermal fluxes.

The solution of Eq. (85) is carried out numerically with the GATHER-II program, modified to include the moments of the thermal source, as they are derived in this section.

5. METHODS OF REDUCING THE PULSE WIDTH

As was discussed in Section 3, one of the principal advantages of the booster over a mechanically pulsed reactor is that advantage can be taken of a poisoned moderator for experiments with low energy neutrons. The poisoning reduces the intensity as well as the pulse width, but it has been shown that the over-all effect is beneficial.

Other means of shortening the neutron pulse also come to mind. One is to vary the pulse by varying the buckling of the moderator. This appears to be inferior to using a neutron poison; to attain the same pulse width by leakage variation as by poisoning, one must accept a lower intensity. This is because leakage tends to reduce intensity at all energies, whereas poisoning reduces intensity primarily at low energies.

With a steady-state reactor, a neutron chopper is used to obtain pulses. The use of a chopper on a pulsed reactor presents an additional problem. If the chopper is not immediately adjacent to the moderator, the neutron beam will be monochromatized by the chopper. For neutron diffraction experiments, in which a white beam is desired, this would be undesirable. On the other hand, placing the chopper deep inside the reactor leads to serious design problems. Instead of using a chopper, then, resolution can be improved by moderator poisoning or by using a long flight path. Of course, a chopper can easily be used to reduce the background between pulses, but not to shape the pulse.

6. REACTIVITY VARIATION DEVICES

In a booster, a mechanical device to make a periodic variation of the reactivity is an optional device which serves several purposes. These are:

1. Higher average powers than are feasible with a constant reactivity device are attainable.
2. A higher fraction of the energy can go into the pulse: The fraction of the total energy which goes into the pulse is approximately $\frac{|\rho|}{|\rho| + \beta}$, where ρ is the reactivity between pulses. The rest of the energy is wasted.
3. The lower power which is produced between pulses also produces a lower background flux, which facilitates the performance of experiments.

The presence of the wheel also introduces some problems. These are:

1. A sizable part of the available reactor space must be provided to accommodate the reactivity changing component.
2. Additional precautions must be taken to ensure that the moving component will not be the cause of an accident.

A number of methods of varying the reactivity have been proposed or are being used. In the Russian IBR reactor at the Dubna Institute (6), two wheels are used. One wheel, having a diameter of 1.1 meters, moves within a gap through the middle of the core. This wheel rotates with a frequency of from 2000 to 6000 rpm. The other wheel, smaller than the first, rotates past the outer surface of the reactor with a frequency which is a submultiple of the larger wheel. Each wheel has a disk of U-235 pressed into it. When both U-235 disks are in their most reactive positions, the reactor is above prompt critical; otherwise it is below delayed critical. The maximum reactivity swing is 7.4% δk . The SORA reactor design (7) has one moving component. It consists of a moving block of beryllium which moves past one face of the reactor core. The block is held at the end of an arm, one meter long, which rotates at 3000 rpm. The maximum reactivity swing is 3.2% δk .

Both of these methods of reactivity variation have much merit. For completeness, we have investigated a number of other possibilities in a preliminary manner. One of these approaches, which has some advantages, is as follows. A radial reflector immediately adjacent to the core has a number of void slots in it, comprising about 10 - 15% of the total reflector. Another reflector, this one capable of revolving around the stationary one, is made up of a number of reflector pieces. The moderator is located just outside the moving reflector. As the outer reflector revolves, the reflector pieces of the outer reflector line up periodically with the slots in the inner reflector. This is the moment of highest reactivity. The rest of the time a small fraction of the inner reflector openings are covered by the outer reflector pieces. Most of the moving reflector pieces are then located behind the solid portion of the inner reflector, where their reactivity worth is decreased. One advantage of this approach is that it affords great flexibility in the design. The slots and reflector pieces may be spaced so that they line up once per revolution, or several times per revolution. In this way we can get one pulse per revolution with a short reactivity pulse (about 300 μ sec at a frequency of 6000 rpm), or we can get several pulses per revolution to permit a slower revolution rate for a fixed pulse frequency. Another advantage is that the entire circumference of the reactor is

available for experiments. In the SORA design, about half of the space around the reactor is needed for the moving reflector. The feasibility of this method depends on how much reactivity swing it allows. Preliminary estimates are favorable, and this is now being tested with a two-dimensional transport calculation.

7. A CONCEPTUAL GAS-COOLED REACTOR

A number of heat removal calculations have been performed to assess whether it would be feasible to utilize a gas coolant. There is a considerable inducement to do this to simplify the maintenance problem. For the analysis, the fuel chosen was a plutonium - 10 atom percent aluminum alloy similar to that used in the EBR-1. This fuel was chosen because its thermal properties are well known and because it may be considered an operational fuel material since it has already been used in a reactor. The core was made of 217 Pu-Al rods with a hexagonal cross section of 0.370 in. across the flats of the hex, with a loose-fitting stainless steel hex cross section can 0.010 in. thick and 0.400 in. across the flats on the outside. The resulting gap between the can and fuel is filled with NaK to assure good thermal contact. In this simplified core, provision was not made for the Linac target and any control or safety rods which might be located in an actual core. Thirty kilograms of fuel, with a fuel volume fraction of 0.70 and a gas volume fraction of 0.183 was found to give a critical reactor. For the neutronic calculation, the core was represented as a cylinder 7.54 cm in radius, surrounded by a 4-cm thick tungsten reflector. Outside the reflector was a 7-cm thick water moderator, another 5-cm thick tungsten reflector, and outside of this was located a lead shield. Axially, the reactor was assumed to be unreflected and its height of 16 cm was accounted for by a buckling term. This was done so that one-dimensional neutronic codes could be used. The actual reactor would probably have an axial reflector. With a helium coolant we found that a flow rate of 1800 cubic feet per minute at the core entrance conditions of 100°F and 10 atm. Pressure was more than sufficient to cool 0.5 MW of thermal power. This represents a power density of .185 MW/liter, which is not excessive. Assuming the helium gas to be at atmospheric temperature and pressure at the entrance to the compressor, a compressor power of 320 horsepower was required. The gas passage in the core between fuel elements is 0.021 in., and the calculated pressure drop was computed to be 23 psi. The maximum fuel element center temperature was 820°F, and the average surface temperature was computed to be 370°F. The temperature outside the stainless steel clad was 355°F and the coolant temperature rise in the core was 138°F.

The possibility of using CO_2 as a coolant was also looked at. CO_2 is cheaper, requires less pumping power, and does not have the same leakage problems as helium. For the same geometry, temperature distribution, and heat removal rate, it was found that with CO_2 coolant, the flow rate was $1000 \text{ ft}^3/\text{min}$, the compressor power requirement was 170 horsepower, the pressure drop through the core was about 80 psi, and the coolant temperature rise through the core was about 116°F . From this comparison it appears that it is just as well to use CO_2 for our application. The one consideration to keep in mind is that CO_2 is not good at very high temperatures because of disassociation and oxidation. In the present example, the Pu-Al alloy fuel has a rather strict temperature limitation anyway, so this fuel is well suited to CO_2 cooling.

The Pu-Al reactor has good neutronic properties. Its prompt neutron lifetime is 1.6×10^{-8} sec. A time-averaged thermal flux of $3.4 \times 10^{13} \text{ n/cm}^2 \text{ sec}$ was obtained in the moderator at an average power level of 1 MW. Taking into account the duty cycle, peak fluxes of about $2 \times 10^{15} \text{ n/cm}^2 \text{ sec}$ in an unpoisoned moderator are obtainable from a reactor operating at an average power level of 0.5 MW.

We have also looked at sodium and the NaK-cooled reactors, primarily for higher power applications. It is feasible to cool a Pu-fueled 5-MW reactor, having a lifetime of about 2×10^{-8} sec using present technology. But it is felt that with the present source strength of Linacs, it would be premature to build a booster of this capacity.

Figure 3 is an artist's conception of a preliminary design developed by C. A. Stevens, J. L. Russell, and G. K. Houghton. It incorporates many of the ideas discussed in this paper. More detailed thermal and neutronic computations on this reactor configuration are presently under way.

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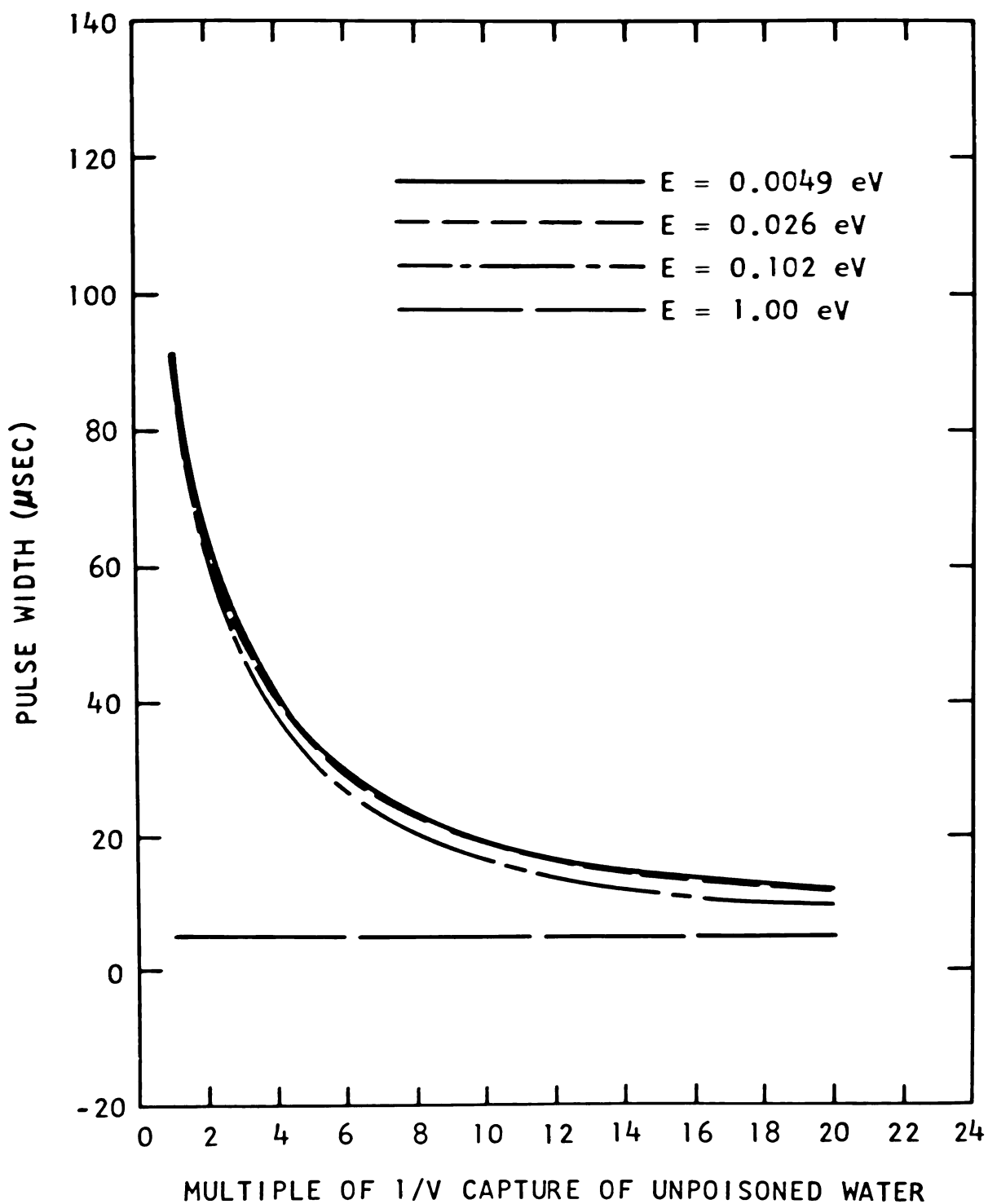


Fig. 1--Pulse width for varying concentrations of $1/v$ poison in moderator of $B^2 = 0.1755 \text{ cm}^{-2}$

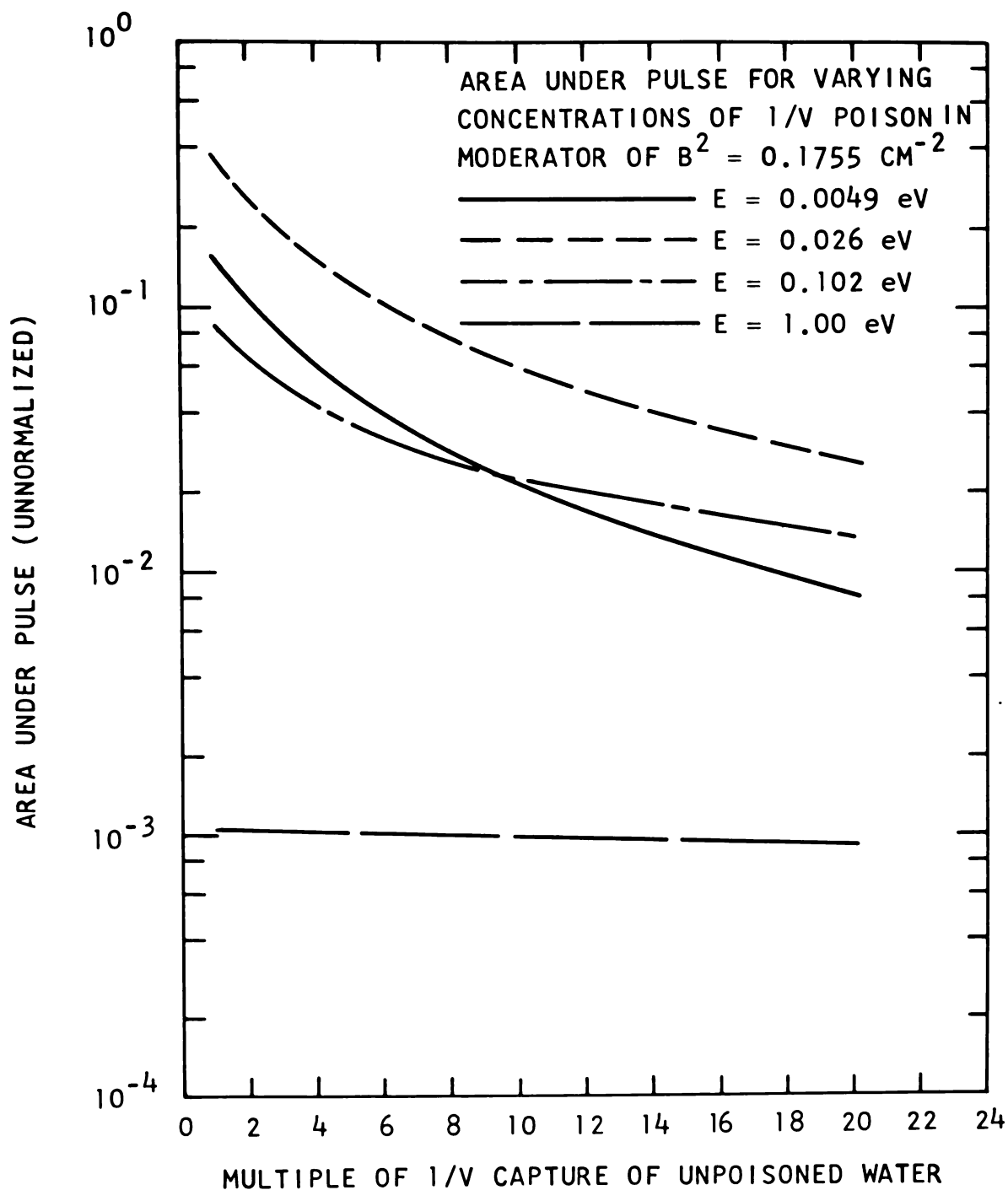


Fig. 2--Area under pulse for varying concentrations of $1/v$ poison in moderator of $B^2 = 0.1755 \text{ cm}^{-2}$

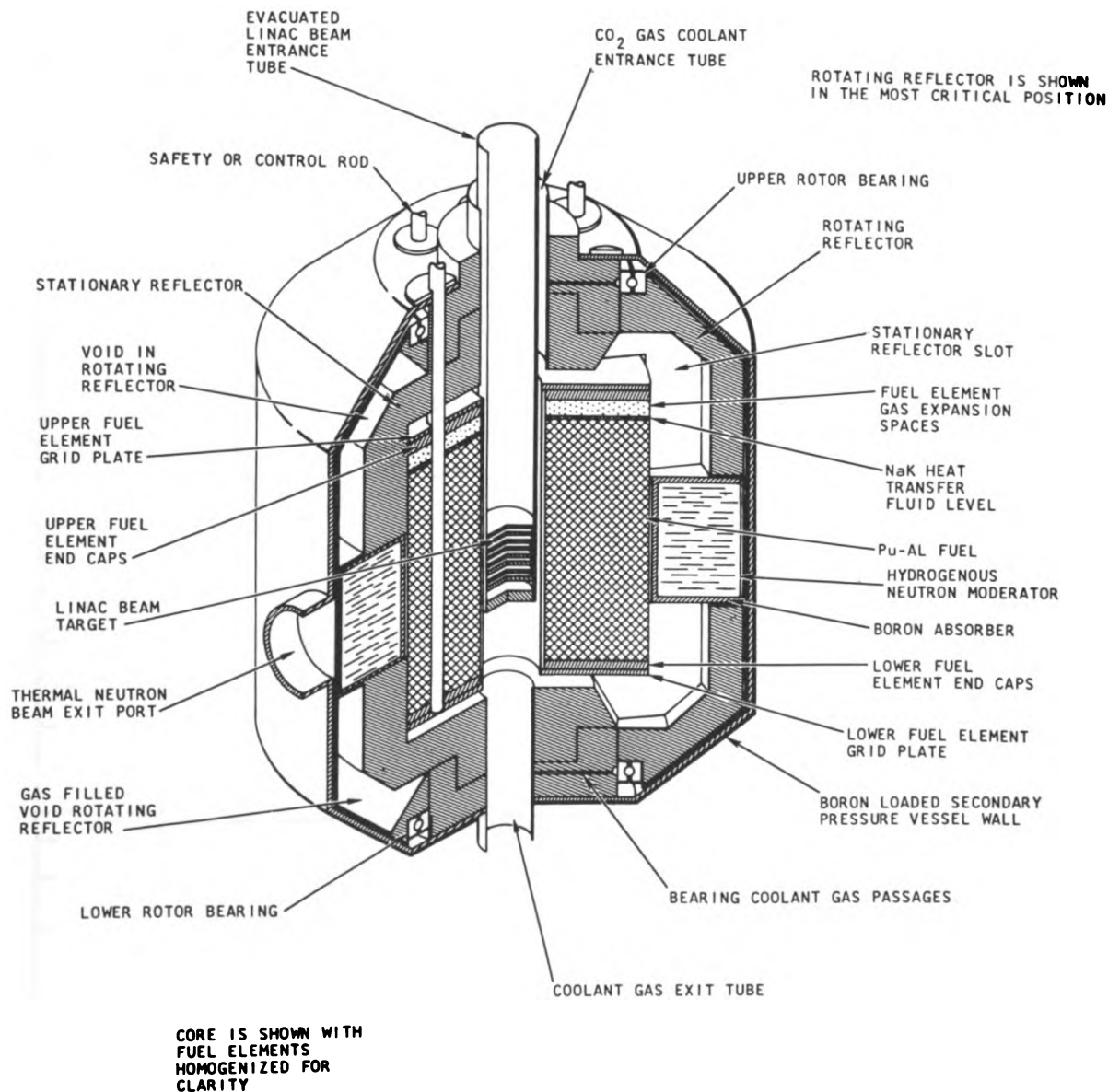


Fig. 3--Conceptual design of an accelerator-booster repetitively pulsed research reactor

DISCUSSION

OF

PAPERS III.A.3. (Poole) and III.A.4. (Stevens)

Chairman: V. Ralevski

Secretary: T. Wimett

CONNOR: Please compare capital and operating costs for the super-booster and the UK high flux reactor.

CROCKER: I would say that the capital costs are probably comparable. For the high flux beam reactor we were very certain that the costs we were quoting were right; we weren't extending the normal engineering standards for continuous reactors very far; the power densities contemplated are quite within limits, and we were sure that we could obtain these fluxes and powers. For the booster, however, there were development costs to which one couldn't really assign definite figures. (For example, the targets and the study of the pulse stresses in the fuel elements.) We, and I think a lot of other people, came to the conclusion that actually the booster would come out a bit cheaper than the continuous reactor. What dictates cost of a continuous reactor finally is the operating power and 100 MW of power requires 30-40 kg ^{235}U a year, which would lead me to think that the booster concept was probably a bit cheaper. The maintenance of the linac and the rotor replacement would be significant. We weren't quite sure of replacement costs for the klystrons, modulators, etc., which one would have to replace on the linac.

CONNOR: Would you comment on the difficulty of synchronization of choppers for particular experiments with a reactivity modulator?

CROCKER: This applies, I should think, with all the rotating devices, but our engineers were pretty confident that we could lock the pulsing system of reactivity in with the rotors and achieve adequate synchronization up to the user requirements.

CONNOR: I want to clarify my question concerning the problem of using an experimental chopper with a booster or pulsed reactor whose timing is controlled by a mechanical modulator. This requires synchronizing two independent rotating devices to within a few μsec . The only solution in present technology is that used in phased chopper time-of-flight systems, e.g., Brugger's at MTR and Eglestaff's at Harwell. I repeat, the only solution as far as I know which exists presently is to run all rotating devices at absolutely constant frequency and with very close limits indeed even on phase. This problem is simply stated but perhaps not simply solved and of considerable importance with respect to the use of pulsed reactors and

boosters.

KLEY: I think we have successfully spun rotors and rotating crystal spectrometers and choppers in phase condition and use an entirely different technique. We simply use dc motors and on one we have an eddy current magnet and use variable friction. With this simple technique, we can phase two systems to within an accuracy of 1 μ sec. We have successfully done this with our rotating crystal spectrometer.

CONNOR: I do not propose to argue with Kley but it does inspire me to define the problem a little more closely. The technical problem is the rate of change of phase which is achievable with a given control system. If the reactor modulator departs from a linear variation of phase with time at some given ramp, then the question is, can the chopper phase be varied sufficiently rapidly? This is a matter of how much power can be put into or taken out of such a system. I am very interested to know that it can be done without difficulty using a break as a source of power.

deBOISBLANC: As long as this appears to be an important technical consideration, I think that one should recognize that the phase requirements are not so stringent upon the large two-rotor system. The phase requirements are not stringent upon the large reactivity varying element so one can concentrate his attention on the small reactivity varying element which has two things which make it possible to get a given precision of synchronization. One is that it is rotating at least twice as fast, so the number of electrical degrees is multiplied by two, hence the phase angle accuracy requirement is halved. Secondly, while you consider the requirements on the control system in terms of rate of change of phase, you also must recognize that the other approach to the problem is to minimize the rate of change of phase that can be introduced by all frictional losses, and that may make a smaller demand upon the corrective network. Now the hysteresis technique which Kley mentioned is obviously satisfactory because the phase loss per unit of time in the system is sufficiently small.

RAJEVSKI: I do not believe that phasing will be a problem if you are working with pulse reactors because the pulsation device has fairly large inertia. It is a big wheel of about one-ton mass and the phase can change only very slowly. Now the pulse width of the thermal neutrons is of the order of 80 μ sec, so if you want to phase a chopper you have a span of 80 μ sec to phase a 10- μ sec chopper. Therefore, I do not see any basic difficulty to phase the chopper with a pulse reactor.

LARRIMORE: Coolant expulsion and "missile generation" during an explosive accident, mentioned by Crocker, have been considered for SORA and are perhaps good examples of how such worries are reduced by detailed design. The effect is reduced by a small coolant volume, namely, 1 liter, in the SORA core and by special arrangements (ring flanges) to prevent missile generation in the beam channels. In addition, the

pressure build-up will occur in the pulsation cell, and will be relieved by openings such as around the top shield plugs as well as the beam tubes. Between the core and the beam tubes there are several layers of material which should attenuate the shock waves.

CROCKER: I'm sure Larrimore has gone into a lot more detail, and I think actually Randalls has discussed this with us in the last year or two. I think we would have concentrated on minimizing the amount of sodium in our system to decrease this, but certainly I think that, without doing more work on control of possible explosions, we could have probably had a shock wave coming down the system. With the moderators we were thinking of, this could lead to mixing of water and sodium.

KLEY: I understood from your drawings that you had only one cooling loop, so that the target is really incorporated into the core. This means that your target is cooled by a light liquid which is sodium. This normally gives a very low yield of neutrons. If you have only a one-loop cooling system, how much do you lose in neutrons, instead of using a two-loop cooling system where you use mercury for target cooling?

CROCKER: I think we did look at mercury as a coolant and it did not accommodate the power density and flows we want to get. Mercury, itself, is a nasty material. The mixture of mercury, sodium, water, plutonium and everything else makes the thing just a little complicated. We felt very convinced that you had to have the same coolant in the target and the core. I don't think that we could have increased neutron yield without having to take out more power and making the system, we felt, more complicated.

KLEY: We came to a different conclusion here, namely that it's better to use a two-loop cooling system because the power density in the target is much higher than in the core. The coolant velocities have to be completely different in the target than in the core itself. So we concluded that it's better to use a heavy liquid metal coolant -- lead or mercury.

CROCKER: As far as I can remember, we just weren't prepared to cool the total core with mercury, and sodium was about our only possibility.

RUSSELL: There is a second factor involved here which is in the reports you refer to. And that is that when you dilute a high Z material such as uranium, with a low Z material such as carbon or sodium, you find that virtually all of the stopping of an electron beam in this dilute material involves interactions with the high Z material. Monte Carlo calculations show that for the very dilute targets that were considered in the Harwell booster, the loss in neutron production over that for a pure uranium target was very small, of the order of 10%, 20% for the most dilute case. This indicates that it is better to cool with a low Z material than a high

Z material which does not have a good neutron production rate. Mercury is one of the poorer neutron producers, compared with uranium.

RAIEVSKI: May I ask of Stevens one question--in the pulsation device, do you rotate fuel or reflector?

STEVENS: In the paper it was stated that the feasibility of the reflector concept depended on how much reactivity could be controlled with it, and, according to calculations done subsequently, it turned out that we couldn't control very much with the reflector at all. So that particular concept has been shelved.

RAIEVSKI: It's a fuel rotor now?

STEVENS: At the moment we're still in the idea stage; we're not building anything and we have no final design.

LARRIMORE: Could you comment on the expected neutron yields from the ^{233}U -W target relative to other materials?

STEVENS: The answer to that is not easy. Part of the program we are carrying out at GA now is the measurement of these yields. The uranium is a better target than tungsten and ^{233}U appears to be better than ^{235}U . But as far as the uranium comparison, I don't think I'm ready to make a final statement about it. Generally, for tungsten the number I used as a rule of thumb was that basically 2% of the electrons are transformed into neutrons.

BEYSTER: There seems to be a little bit of information available on some measurements done previously by Russell and there are indications that either plutonium or ^{233}U will emit something like 1.4 times as many neutrons as ^{238}U per kW of beam power in the target. Also, there appear to be indications that if you increase the electron energy from 30 million volts up to, say 100 or 150 MeV, you will gain another factor of 1.4 per kW deposited in the target. The other advantage, of course, in using the higher energy is the fact that to get the same amount of average power in the beam, you run considerably less current minimizing the target window problem and also the cooling of the target itself because the range is roughly proportional to the energy. There are real advantages to high energy.

BECKURTS: If I understand Stevens, he stated at the end of his talk that for thermal neutron experiments and with a system which had about the same power as SORA, the figure of merit for this proposed booster would be about a factor of two or three higher. Does this gain in figure of merit justify the much increased complexity of the system using a big linac and the much increased cost in investment and operation?

STEVENS: I don't think that the problems of matching the usual linac of today are more complicated than those of a pulsed reactor of the same power and much has been said in the way of simpler mechanical design for the booster, for instance, less

stringent requirements on the motion of wheels and so on. When you get to bigger linacs, we ourselves have not at this stage tried to design a target for one, so I cannot estimate how much cost would be involved. It may be necessary to go to liquid targets, and this question has to be considered more carefully.

KIEY: Stevens remarked that the booster is about three times better for neutron experiments. I think I can agree as far as the assumptions in the analysis are concerned, which refer only to experiments with good energy resolution and not good angular resolution. I hope I have the time during the panel discussion to discuss the few experiments where this is the case; normally you have to consider both energy and momentum, or angular resolution. This leads to a somewhat different conclusion and different results, which means that you don't need much shorter pulses; you can use pulses as they are produced by the pulsed reactor. Secondly, you have another possibility of handling this shortening of the pulses by using the neutron guide tubes of Meier-Leibnitz and Springer. You can lengthen the flight path without losing intensity. Then you will find that your figure of merit ratio is not a factor of three but maybe even slightly less than one.

STEVENS: Experimenters tell me that there is, in fact, a large class of experiments in which energy resolution is of principal importance. These include inelastic scattering, diffraction measurements, and cross section measurements. Dispersion relation measurements do, in fact, require momentum resolution as well. I think there is a wide class of experiments in which energy resolution is all you do care about.

COMPARISON OF REPETITIVELY PULSED DEVICES

V. Raievski

Summary

The main design characteristics of various pulsed systems are reviewed with the objective of comparing and contrasting the operation of ordinary pulsed reactors and pulsed reactors with source injection from a Lineac. A survey of existing proposals is made with special reference to the one system (the Russian IBR) already in operation.

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Introduction

The physics of repetitively pulsed reactors is now well understood and the six years of successful operation of the IBR⁽¹⁾ have demonstrated their feasibility and reliability. It has also demonstrated the great interest of these devices as a source of neutrons for nuclear and solid state physics. The time is now appropriate to go to larger units.

The different designs under consideration range in power from tens of kilowatts to tens of megawatts, but only one device of this type is now in operation and has a power of 6 kW. This fact reflects either the optimism of the present designers or the conservatism of the original IBR design which was intended to run at only 1 kW. Personally, I tend to follow the optimistic mood. At high power operation, these devices are capable of satisfying a wide range of needs extending all the way from the basic training of physicists in the nuclear and solid state field up to the most advanced research in universities or research centers.

The different repetitively pulsed neutron sources

Neutron pulses can be obtained either by reactivity or source variations in a multiplying assembly or by the combination of these two methods. Thus there are three types of system to consider. The simplest, from the engineering point of view, is source injection without reactivity variation, a method used at Harwell since 1958. This device, developed by Poole, is known as Harwell Booster⁽²⁾. Source neutrons are generated in the assembly by electron bombardment from a lineac.

The lineac is not a very efficient way to produce neutrons since it requires about 15 times more energy per neutron than in a fissile assembly. Thus, the power density in the electron target is very large (3 MW/liter for the lineac of the Harwell booster). The main point, however, is that the neutron source can be made of very short duration to give a high energy resolution for a given flight path.

Modern lineacs are operating now with a mean power of 50 kW (e.g. the RPI), and a ten times larger power seems

feasible in the future if further technical development is undertaken. However, highly concentrated fast reactor cores are currently operated at power densities around 1 MW/lit and total powers around 60 MW. Thus it is clear that a lineac cannot compete with a reactor as a neutron source except for those areas where very short neutron pulses are necessary. This means that there is a strong incentive to extend the use of pulsed multiplying assemblies for experiments requiring low energy high flux neutron sources where the pulse width is broadened by the slowing down and thermalization process. This incentive is apparent in the use of the IBR and is reflected in all the new designs.

Pulsed reactors and accelerator pulsed reactors*

In this report I shall refer to multiplying assemblies with modulated reactivity as "Pulsed Reactors", and those with modulated reactivity and synchronized source injection as "Accelerator Pulsed Reactors".

Since pulsed systems can be operated at high power, high amplification ratio is required in a source injection system, where the target neutron yield is inherently limited. The only way to produce a strong power pulse is to operate the assembly near to prompt critical, but in this case a source injection system suffers from a bad concentration of power in the pulse and a consequently high background. This is a consequence of the multiplication of the

*The following abbreviations will be used in this report:

PR Pulsed Reactors

APR Accelerator Pulsed Reactors

delayed neutrons produced during the pulses and could be improved either by decreasing the number of delayed neutrons present in the core or by decreasing the reactivity between the pulses. The first possibility could be achieved, in principle, by employing a circulating liquid fuel which automatically removes the neutron emitters, a method proposed by Fluharty⁽³⁾, but never applied. We will come back to this proposal later on. The second possibility, which is the one realized in the IBR and used in all the proposed designs, can be achieved by using a mechanical device (e.g. a moving reflector) which depresses

the reactivity between the pulses.

The pulsation mechanism

The necessity for a reactivity modulation is easily seen from the formula relating the energy in the pulse to the mean energy:

$$\frac{P}{\bar{P}} = 1 - \frac{\beta}{\epsilon_0} \quad (1)$$

where ϵ_0 is the absolute value of the prompt reactivity between the pulses, and β/ϵ_0 is the background to mean power ratio.

The relation is valid either for a pulsed reactor or a booster, but for a booster we have, in addition, the following relations for the amplification and relaxation time as a function of the modulus of the prompt reactivity, ϵ_1 , during the pulse:

$$A = \frac{1}{\epsilon_1}$$
$$\theta = A \tau$$

It is obvious from these relations that a high amplification factor means a large relaxation time and a large background to mean power ratio with a correspondingly poor concentration of power in the pulses.

There is no way to change the relation between the amplification factor and the relaxation time, but there is a possibility of achieving a large amplification with a small background to mean power ratio if the prompt reactivity is made a periodic function of time, $\epsilon(t)$, whose modulus is small during the pulse and large in between. This is the principle on which all designs are based. Thus a necessary condition for good pulse characteristics is

a large reactivity worth for the pulsation device.

For a high amplification factor, the reactivity would oscillate between the value $\epsilon_1 = \beta$ and $\epsilon_0 = K\beta$, where K is a parameter which measures the quality of the pulsation device.

Now there is an advantage to operate an APR with the highest possible repetition rate. For a given peak power of the accelerator this would give the maximum counting rate with the shortest pulse. The maximum repetition rate is determined by the pulsation device. It is not easy to achieve a large K value

since this necessitates the movement of large parts of the reactor. Let us assume tentatively that the moving part with its holding items has linear dimensions of the order of 0.1 m. This implies that in order to form a complete pulse, the part must move during one cycle over a distance of at least the order of 0.2 m. Thus, for a repetition rate even as high as 250 cps, the speed of the moving part would be of the order of 50 m/sec. Such a low speed would inevitably lead to very broadened reactivity variations looking more sinusoidal than pulse like. It may be demonstrated that for a sinusoidal variation, the reactivity which enters the formula (1) for the ratio of the energy in the pulse to the mean energy is the geometrical average of the minimum and maximum values of the reactivity:

$$\overline{\epsilon}_0 = \sqrt{\epsilon_{\min} \epsilon_{\max}} = \beta \sqrt{K}$$

Such a device would be very inefficient. There are, however, two ways of increasing its efficiency:

- a) to increase K by increasing the dimensions of the moving part, a solution greatly limited by the dimensions of the core; or
- b) to increase the distance between two consecutive reactivity insertions in order to get a reactivity variation near to a true pulse shape with its K dependence rather than a sinusoidal shape with its \sqrt{K} dependence. In both cases the tip velocity would be increased above the 50 m/sec, but the last method is easier to achieve. This means that the mechanical device would look like a large wheel rotating at high velocity.

For instance, the IBR wheel has a diameter of 1,1 m and a tip velocity of 230 m/sec. Similar values are used in all the other proposed designs with only one exception. The SORA wheel has a diameter of 1,8 m with a tip velocity of 283 m/sec. The BNL advanced booster has a wheel of 1,27 m diameter with a tip velocity of 220 m/sec, the advanced pulsed reactor has the same wheel rotating at a tip velocity of 265 m/sec. For the Harwell super booster it is proposed to follow the SORA design for the pulsation device. In contrast, the proposal of the GA accelerator booster is rather different, but it must be demonstrated that it can operate both from the engineering and the reactor physics point of view.

The pulsation device used in the IBR has proved to operate satisfactorily at a reactor power of up to 6 kW. It is now planned⁽⁴⁾ to operate this reactor in 1968 at 30 kW power. The limiting factors in going to this power lie in the pulsation device where the allowed thermomechanical stresses on the fissile U-235 sample fixed on the pulsation wheel are exceeded at 15 kW core power. To achieve the 30 kW power goal, two U-235 inserts would be used, thus dividing the power evenly between the two inserts. It is clear that for higher power, other means must be found for pulsing the reactivity. This is achieved by the SORA proposal for pulsing the reactivity by moving a part of the reflector, which has been proved to operate in an extensive program including engineering tests and reactivity measurement on a nuclear mock up. This is a result of importance for the development of pulsed systems and it demonstrated that pulsation devices for reactor powers of tens of MW are conceivable on the basis of conventional techniques.

To summarize this point, I would say that in all designs it is proposed to use a movable reflector for producing the periodic reactivity variations (**except the Triga system which will probably use a moving poison**), that, with the exception of the GA accelerator booster pulsed reactor, the mechanical movement is achieved by a large wheel rotating at high tip velocity, and that it has been demonstrated in the SORA test program that this system gives the reactivity variation required with good pulse characteristics usable in systems operating at high power (i.e. tens of MW).

Pulsed reactor versus accelerator pulsed reactors

The tendency to use these systems in solid state physics, where already high flux steady state reactors operated with choppers, give high intensity neutron bursts, is an incentive to achieve the highest possible reactor power. Already pulsed systems of a few hundreds of kW give pulse intensities of the range of 10^{15} n/cm²/sec, comparable to HFR intensities, and they can compete with these systems successfully in many fields of solid state physics. But a value of a few hundreds of kW is very conservative and powers up to a few tens of MW are conceivable using present day technology. At these powers the limitation in power of the lineac would necessitate very high

amplification factors such that the relaxation time of the power pulse would exceed the value of interest for the experimenters and there would not be any interest to increase the power further. In this case it is worthwhile to get rid of the accelerator and generate the power pulse with the reactor alone by going to super prompt reactivity during the pulse. This way of operation has been demonstrated by a lot of tests on single burst reactors like GODIVA and is currently used in the IBR. In this case the power burst width is fixed by the neutron generation time in the core and the reactivity insertion rate near prompt criticality of the pulsation device and is independent of the power level. Typical values for fast reactors range from 30 to 60 μ sec 2-3 msec for a thermal reactor like the proposed Triga. This philosophy is well reflected in the BNL proposal. The APR is proposed for a maximum power of 9 MW, which is felt as a limit due to the availability of powerful lineacs. The proposed lineac would have a peak power of 1000 MW and a mean electron beam power of 480 kW. The Harwell APR is rated for the same power of 9 MW and the lineac would have a peak power of 150 MW with a mean electron beam power of 300 kW. It is worthwhile to notice that the best machine now in operation is the RPI with a peak power of 80 MW and a mean electron beam power of 50 kW. Such machines as proposed by BNL and Harwell thus represent an important improvement on the present day technology of lineacs. The power of 9 MW is not a limit for the reactor core, and the PR of BNL is designed for 30 MW power.

The PR necessitates a pulsation device giving not only a large reactivity worth but also a large reactivity insertion rate near prompt criticality. In APR operation, only the first condition is required, but we have seen that the high repetition rate in an APR would lead to the necessity to design the pulsation device in such a way that this second condition would also be realized.

a) Operational conditions:

It is essential in an APR to achieve a mean neutron generation time as low as possible, the pulse half width being directly proportional to this number. Thus a very compact core with a small amount of structural and coolant material must be designed and the reflection of moderated neutrons from the reflector and

scatterers must be carefully avoided. This certainly has a limiting effect on the core power density, but is less important in a PR where the pulse half width is only proportional to the third power of the mean generation time. Thus a PR can be designed and operated at higher power than an APR core.

It is instructive to show in a simplified manner the condition of APR operation. Let us suppose that the pulse duration of the lineac is short compared to the relaxation time of the multiplying assembly. The energy of the reactor pulse is proportional to the energy of the lineac pulse multiplied by the amplification factor:

$$P_M \bar{\theta} = K P_b t_a A \quad (2)$$

where P_b and t_a are respectively the peak and the duration of the lineac power pulse and P_M and $\bar{\theta}$ the same quantities for the reactor power pulse. As the relaxation time is equal to the generation time multiplied by the amplification factor,

$$\bar{\theta} = A \tau$$

one can see that the peak power of the multiplying assembly depends only on the energy in the lineac pulse:

$$P_M = K \frac{P_b t_a}{\tau} \quad (3)$$

This relation shows clearly the importance of the mean generation time.

Thus the peak intensity depends only on the power of the lineac.

Now, if the experiment requires a given value of $\bar{\theta}$, there is a strong incentive to limit $\bar{\theta}$ to this required value, otherwise the pulse must be cut by some other method (phased chopper, poisoning of the moderator or reduction of its dimensions). All such methods are wasteful in neutrons. Reducing $\bar{\theta}$ to the required value means that the repetition rate can be increased to the value where the mean reactor power is equal to the design power according to the relation:

$$P_M \bar{\theta} f = \bar{P} \quad (4)$$

If the value f given by this relation exceeds the maximum repe-

tition rate of the pulsation device or if it must be limited by the frame overlap condition, the reactor would be operated at a power below the design value.

In PR operation, the peak power can be adjusted arbitrarily for a given mean power by changing the pulse repetition rate, as seen in the relation (4). This is a definite advantage for experiments with low repetition rate.

b) Frequency selection:

As different experiments require different repetition rates, the means to change this parameter must be examined in the two systems. In the PR, one method is to use two wheels, one with constant velocity, the other with a variable velocity. The repetition rate is fixed by the beats between the two wheels. The pulse shape which is determined by the constant velocity wheel, is independent of the repetition rate.

In the APR, the variable speed wheel is unnecessary and the repetition rate can be varied by varying the repetition rate of the lineac between zero and the repetition rate of the pulsation device in such a way as to achieve coincidence.

c) Control:

Both the PR and APR systems are reactors and need control and safety rods. The APR is a very stable device and the performance required of the control system is not highly demanding. The peak power stability depends essentially on the electron beam power stability. In contrast the PR is extremely sensitive to reactivity fluctuations. Thus, in the IBR, reactivity variations of 1,4 pcm during a period induce, at a repetition rate of 3,3 cps, a 10 % peak power variation. In SORA at 50 cps, 2,4 pcm are required to give the same peak power variation. 2,4 pcm per period means $0.12 \% \text{ sec}^{-1}$ which is not in fact a small reactivity variation. However, all effects which can produce reactivity variations like mechanical, thermal or hydraulic effects must be carefully reduced by a proper design.

The IBR operation has demonstrated that this can be achieved and the studies made on the SORA design have confirmed that it can be achieved within the standards of present day technology, e.g. the velocity stability of the SORA pulsation device must be better than 3 % to maintain the peak power fluc-

tuation below 10 %. Finally, in the design of a PR all reactivity fluctuations must be kept as low as possible, to maintain a good power stability, and this can be achieved within present day technology.

d) Safety:

The safety problems of these two devices (the thermal PR Triga excepted) are those of all fast reactors with highly concentrated cores. The worst imaginable circumstances are those where the core is forced into a prompt critical assembly. In such a hypothetical accident, a certain amount of energy is generated before the bulk expansion of the core is able to terminate the nuclear excursion.

A collapse of the core can be caused by a number of circumstances but the worst hypothetical accident depends on the particular reactor system. In such system where the pulsation device is a heavy moving part, as in the SORA reactor, the worst hypothetical accident takes the form of a fracture of the device during operation and the collision of the broken part with the core in such a way as to suddenly compress the core into a highly supercritical assembly. Our studies in the SORA project have concluded that for a certain model of this accident (which is periodically subject to improvement), the maximum energy release is about 170 kgm TNT.

On the other hand, in a system where the pulsation device is not very large (if such one ever exists), the worst hypothetical accident takes the form of a fuel melting accident (due to a loss of coolant for example) and the formation of a supercritical assembly due to the compaction of the molten core under gravity. The Harwell superbooster team have considered such an accident in their preliminary design and estimated that for an assumed reactivity ramp input rate of 100\$%/sec (due to the collapse of the core under gravity) the destructive energy release would be about 30 kgm TNT.

The above two types of accident are not the only extreme accidents which have to be considered in PR's and APR's. There exists also the just-credible accident of a rapid drift in the base reactivity of the system during otherwise normal pulsed operation. In SORA calculations, it has been estimated that fuel vaporization will not occur below about 5\$%/sec and that above

this drift rate the destructive energy release rises to a saturated value of about 100 kgm TNT at a drift rate of 50\$/sec.

This saturation effect is very important since it serves to reduce the apparent difference between the safety features of PR's and APR's. All other things being equal, this difference lies in the fact that an APR is operated at a peak reactivity of about 100 pcm below a PR. Hence, the saturated value of the energy release occurring at large reactivity drift rates is the same in both cases.

In conclusion, the safety problem for PR and APR operation are quite similar. An important point is the saturation limit of the energy released in the worst hypothetical accident, which from the calculations on the SORA reactor, appears to be considerably less than normally expected in a fast power reactor. Another important fact emerging from the SORA studies is that feedback mechanisms such as the Doppler effect and the axial fuel expansion have only a small effect on nuclear excursion and can be ignored.

Comparison between the different designs

Some of the characteristics of the different systems are reproduced on Table I.

⁽⁴⁾
IBR: The IBR has been in operation at Dubna since 1960. It is the first repetitively pulsed reactor and has demonstrated the reliability of the concept and its value for physics research. It is a very simple device, cooled by air flow at atmospheric pressure and unshielded. The reactivity pulse is produced by the periodic insertion of a U-235 sample. The power was first rated at 1 kW and increased to 3 kW in 1964 and 6 kW in 1965, at which power it is still operating at the present time. The improvement was achieved merely by increasing the air flow rate that cools the stationary core and providing cooling to the emergency rods. A new operation procedure called "alternating pulse power operation" was also introduced to take advantage of the special features of a PR for certain experiments requiring a very high signal to background ratio.

The practical way to achieve this low repetition rate with the existing equipment was to rotate the auxiliary wheel at a very low speed, 0.12 cps. The rotation of the principal wheel

(50 cps) then generates a group of pulses, this pattern being repeated each .12 sec. In this mode of operation, the power of central pulses in the group is approximately 7 times higher, at a mean reactor power of 4 kW, than for usual operation. This is not the best way of reducing the pulse repetition rate and further improvement is now being investigated in order to get a single pulse, instead of a group of pulses at a very low repetition rate.

Another improvement introduced at the end of 1964 for spectroscopic measurements was the reduction of the pulse width by coupling a microtron with the reactor. Because of the limitation in the microtron power, the APR operation conditions are limited to a mean power of 1 kW and a constant pulse rate of 50 cps.

It is currently planned to increase the mean power in PR operation up to 30 kW, where the limitation lies in the thermo-mechanical stresses in the U-235 insert. This problem will be solved by using two U-235 inserts in the pulsation device so as to divide the heating in each by two. It is worthwhile to notice that operation at this mean power and 3,3 cps would give thermal cycling on the fuel element equal in amplitude to those arising in a PR operating at 50 cps and about 500 kW mean power and would provide a valuable test of fuel elements operating in those very peculiar conditions.

SORA:⁽⁵⁾ Among the proposed designs, the Ispra design SORA is in the more advanced stage and has been checked in an extensive program including thermal, mechanical and reactor physics tests. It is the first design which includes a liquid metal coolant and a reactivity pulse produced by the periodic movement of part of the reflector. This seems to be the only practical way to pulse a reactor of high mean power. The actual experiment performed on the SORA mock up at the Oak Ridge Center has demonstrated that reactivity pulses of up to 6 % amplitude having the required rate of insertion can be achieved in a practical design. In this experiment, the "window", that is to say the region between the core and the moving reflector, was mocked up by 5 mm Fe/ 2 mm air/ 3 mm Fe/ 3 mm Al/ 4 mm air, which is quite representative of a practical design part of a high power pulsed reactor.

The SORA has included all the features of a high power pulsed reactor. As stated earlier, it is interesting to push a PR de-

sign up to the limits of technology. Thus it might be asked why the SORA design is so conservative on power. There are a few reasons for this choice. The first one is historical. At the beginning it was felt that a power increase of about a factor 100 above the IBR operating condition (at that moment 1 kW) was an important step. It was gradually realized during the studies that this was too conservative and the power was increased step-by-step from 100 kW to 1 MW. At this point it was felt that the cost of the device must be limited anyway to a certain value and it was decided to fix the power at 1 MW and make a full design with a thorough cost evaluation. Hence, this limit is by no means a technical one and may depend on the degree of optimism of each designer as regards the generosity of the financiers.

The second reason is related to safety. It was decided to design the core for a Pu fuel but to operate the reactor first with a U-235 fuel. To achieve the highly concentrated core required, metallic fuel was necessary and it would be difficult to reach higher specific power in such a concentrated core with this fuel.

The third reason is an "a posteriori reason". It was realized that a 1 MW power was not a bad choice. The design of every component of the reactor uses conventional technology. No extensive technological research and development program is necessary. The coupling of a lineac with the reactor can wait until such time as it becomes commercially available and fully tested.

Finally, it has been established by the program of physics studies on SORA that a mean power of 1 MW will give a performance which competes favourably in many fields with a steady state high flux reactor while costing considerably less.

It should be noted that the very good performance with a U core would give optimum pulse characteristics and background conditions when operated as PR with a Pu core.

In any case, the SORA reactor would be an important and maybe necessary step towards high power PR or APR.

The super booster (6)

All the experience gained by eight years of operation of the Harwell booster has been incorporated in this design. This

is particularly important since the most critical points in the design are probably the accelerator and especially the target and target window region. According to the SORA tests, the window between the core and the pulsation device does not seem to be a particularly difficult part. An important result of this design is that 9 MW power can be reached in a Pu ceramic core of 6 liters. However, there is no evidence that the Pu fuel element would sustain the thermomechanical stresses which would be generated at low repetition rate. The safety features of a super booster have been thoroughly examined and it has been demonstrated in a model explosion experiment that a building could be constructed to contain even the worst hypothetical accident. This design also shows that it would be possible to accomodate up to 20 good beam tubes, even in such a small volume core.

It is important to note that though the accelerator is a significant extrapolation of existing lineacs, its performance is not sufficient to take advantage of the full reactor power and that the reactor pulse width must be increased up to 20 μ sec or the reactor mean power decreased to 2 MW for certain experiments.

The advanced booster⁽⁷⁾

This system/^{under} investigation at the Brookhaven National Laboratory, has characteristics similar to the Harwell super booster, but the proposed lineac is a still more powerful machine.

The advanced booster would deliver pulses as short as 6 μ sec even at full power. As for the Harwell super booster, the problem of the accelerator, target and target window would be particularly important and it seems that a research and development program must be undertaken to demonstrate that they can be solved.

It seems that this is the most advanced design which can be conceived within the context of present day technology.

The advanced pulsed reactor⁽⁷⁾

This second Brookhaven design clearly adopts a similar philosophy as that advocated in this report. As the power attainable by an advanced booster is restricted by the inherent limitations associated even with the large lineacs of the future, there is a very strong incentive to get rid of the lineac and

design a PR at the maximum power on the basis of existing technology. This maximum power was assumed to be 30 MW for a 20 liter core containing solid fuel. The core power density of 1,5 MW/liter is still of the same order as that for fast power reactors but heat extraction is easier for a research reactor as the coolant temperature is lower.

The flux available would be higher than for any existing system. However, the excess of the flux over that for a 1 MW reactor would be less than the power ratio because of the larger volume of the core.

A particular uncertainty in this design is the lower limit of the pulse repetition rate. This was tentatively fixed at 22 cps but it is not proved that the thermomechanical stresses on the fuel element would permit operation at such a low level. If not, then this limit must be raised to accord with the fuel element test results.

The philosophy on which the BNL studies, both the PR and APR, is based is that of pushing the designs to the limit of present day technology and specializing each device in a certain field of research. It is clear, however, that these devices cannot be built before a significant research and development program is performed.

The accelerator booster pulsed research reactors (8)

This is an outgrowth of a program at the General Atomic center at San Diego, to study the production of single, very high intensity pulses with controllable pulse width.

The proposed design differs from the preceeding one in the use of a gas coolant (pressurized CO₂ or He), thus being closer to the IBR concept than a system cooled by liquid metal. In addition, the mechanical design of the pulsation device is different, although the principle of reflector pulsing is conserved.

The use of gas cooling limits the attainable mean power and a value of 0,5 MW was chosen as a reasonable limit. Thus this device must be thought of as the ultimate in gas cooled pulsed reactors and not as the prototype of a higher power system.

The advantage of gas cooling is that it is a well known and proved technology.

A large effort has been devoted to investigating the moderation process in the homogeneous source, but further work is still necessary before definite conclusions can be drawn.

The proposed pulsation device embodies many attractive features but a lot more engineering and reactor physics studies will be necessary to demonstrate that it can operate and give the required performance.

The multiple pulsed Triga ⁽⁹⁾

This is an extension of the well known Triga reactor developed by General Atomic and established as a safe and reliable system. With a mean power of up to 10 MW, which seems a reasonable extrapolation over the present Triga reactor power, the thermal flux in the beam tubes would be of the order of 10^{15} n/cm² sec at a repetition rate of 50 cps. The pulse width would be 2 - 3 msec.

For some experiments the large value of the pulse width is acceptable but for many others the width must be reduced, for instance by employing a phased chopper. This is a method wasteful in neutrons.

However this would provide very high pulsed neutron fluxes, using a very safe thermal reactor with a well known technological base, a fact of particular importance for universities.

Conclusion

There are now many designs of high power pulsed assemblies. The prototype of these has successfully and reliably been operated for six years and its power has been increased step-by-step from 1 to 6 kW and is now scheduled for 30 kW.

The designs, supported by engineering and reactor physics tests, show that such a device can be built and operated at a power of tens of MW, giving very high neutron fluxes with pulse characteristics suitable for a large field of neutron physics experiments, including the physics of condensed matter and nuclear physics.

Most of these designs use present day technology and a few rely on reasonable extrapolations of technological trends. All of them are far from the basic limitation of the pulse concept. This becomes apparent if we imagine a system (at present unattainable) using a circulating liquid fuel. Such a system

Name	Lab	Mode of Operation	Reactor Mean Power	Coolant	Pulse repetition Rate cps	Accelerator Characteristics	Peak Beam Power	Mean Beam Power
IBR	Dubna, USSR	PR	1 kW (1964) 3 kW (1965) 6 kW (1965) 30kW(planned for 1968)	air cooled at atmospheric pressure	3,3 - 83			
IBR	Dubna, USSR	APR	1 kW (1964)	"	50		2 MW	200 W
SORA	Ispra, Euratom	PR	600 kW	NaK	50			
SORA	"	APR	1 MW	Nak	variable up to 150		50 MW	45 kW
Super-Booster	Harwell, UK	APR	9 MW	Na or NaK	up to 200		150 MW	300 kW
507 Accelerator booster pulsed reactor	San Diego GA USA	APR	500 kW	He 10 atm or CO ₂	-		-	-
Advanced Pulsed Reactor	BNL, USA	PR	30 MW	Na	22 or more			
Advanced Booster	BNL, USA	APR	9 MW	Na	up to 160		1000 MW	480 kW
MP Triga	San Diego GA USA	PR	10 MW	H ₂ O	up to 50			

TABLE I

* PR abbreviation for Pulsed Reactor

APR abbreviation for Accelerator Pulsed Reactor

would be capable of giving neutron fluxes of the order of 10^{18} neutrons/cm² sec with a repetition rate of 50 - 100 cps at a power comparable to present day power reactors, thus opening completely new fields of research.

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ENGINEERING PROBLEMS IN PULSED
NEUTRON SOURCES*

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SUMMARY: Reference design studies have been accomplished for two high performance repetitively pulsed neutron sources: an advanced pulsed fast reactor, and an advanced accelerator-booster system. The major engineering problems associated with each type of device are discussed along with a description of the method of analysis and a summary of the analytical results.

The Advanced Pulsed Reactor studied has an average power of 30 MW in a 20 liter core producing 1.1 MW-sec bursts of energy with a repetition rate of 22 bursts per second. The total leakage neutron yield per burst is 5×10^{16} with a burst width at half power of 60 microseconds.

The Advanced Booster concept has an average power of 9 MW in a 6 liter core producing 160 bursts per second with .044 MW-sec of energy in each burst. The core assembly has a prompt multiplication of 250 and is driven by an electron accelerator that develops a 33 amp average current of 30 MeV electrons for 3 microseconds. The total leakage neutron yield is 2×10^{15} neutrons per burst with a burst width of half power of 6.5 microseconds.

The thermal, hydraulic, mechanical, and physics analyses of both systems are presented. The major engineering problems are encountered in the fuel elements and in the rotating assemblies containing fuel and reflector pieces for reactivity pulsing. The performance requirements for these components are

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evaluated and discussed. The study indicates a special need for fatigue data and information on other mechanical properties of fuel and cladding materials.

1. INTRODUCTION

A study has been made of advanced research reactor concepts to provide a very high peak neutron flux for pulsed neutron beam experiments. Analyses have been made of an advanced repetitively pulsed fast reactor and a repetitively pulsed fast assembly driven by an electron accelerator neutron source (Advanced Booster). The thermal, hydraulic, mechanical, and physics aspects of these systems are outlined along with an evaluation of the major engineering problems that are encountered.

The relative characteristics of several existing and proposed pulsed neutron sources are illustrated in Figure 1. The full burst width at half power is plotted against the total number of leakage neutrons in a burst. The peak neutron intensity, in leakage neutrons per second, is indicated by the diagonal lines and the total leakage neutron yield in a burst is related to the average reactor power, as indicated, for the operating conditions of 50 bursts per second, a heat removal requirement of 133 MeV per leakage neutron, and with 80% of the energy generated in the pulse.

After a survey of the state of the art in pulsed devices, design objectives were established for each system under consideration.

The objective for the Advanced Pulsed Reactor is essentially a compromise between the SORA design¹ scaled up in power and the Oak Ridge HPPR^{2,3} scaled down with respect to pulse energy to a value that is reasonable for a highly repetitive system. To maintain a reasonable prompt neutron lifetime and pulse width, the core volume is limited to 20 liters. A conceptual layout of the system is illustrated in Figures 2 and

3.

The characteristics of the advanced booster system are determined by an estimate of the performance of a scaled-up standing wave linac of the PHERMEX type⁴.

The experimental aspects of the two systems cannot be easily compared since each would undoubtedly be optimized differently for various types of experiments.

2. SUMMARY OF DESIGN CONDITIONS

The design conditions are summarized in Table I for both the Advanced Pulsed Reactor and Advanced Booster.

TABLE I

Summary of Design Conditions

	<u>Advanced Pulsed Reactor</u>	<u>Advanced Booster</u>
<u>General</u>		
Core size, liters	20	6
Average power, MW	30	9
Energy in pulse, MW-sec	1.1	.044
Repetition rate, sec ⁻¹	22	160
Peak power, MW	18,300	6,800
Pulse width at half power, sec	60 x 10 ⁻⁶	6.5 x 10 ⁻⁶
Total leakage neutrons per burst	5 x 10 ¹⁶	2 x 10 ¹⁵
<u>Physics</u>		
Fuel loading, Kg U ²³³	78	41
Prompt neutron life- time, sec	4 x 10 ⁻⁸	2 x 10 ⁻⁸
Prompt multiplication	-	250
Number of days to 1 a/o burnup	26	46

TABLE I
(continued)

	<u>Advanced Pulsed Reactor</u>	<u>Advanced Booster</u>
<u>Mechanical</u>		
Fuel Elements		
Element O.D., inches	3/16	3/16
Na bond thickness, inches	.005	.005
Cladding thickness, inches	.015	.015
Fuel Pin Diameter, inches	.1475	.1475
Pin spacing, P/D ratio	1.1	1.1
Fuel Material	U^{233} alloy	U^{233} alloy
Fraction of Diluent		
in fuel, v/o	56 v/o	22 v/o
Number of elements in core	3,100	1,385
Radius of rotating		
wheels, inches	40	40
Speed of primary reflec-		
tor wheel, rpm	2640 rpm	2400 rpm (4 arms)
Speed of secondary		
wheel, rpm	1320 rpm	
<u>Thermal and Hydraulic</u>		
Coolant	Na	Na
Inlet temperature, °F	400	400
Bulk ΔT , °F	278	187
ΔT in fuel, Average, °F	294	295
ΔT in Na bond, Aver., °F	7.2	7.2
Film ΔT , Average, °F	34	34
ΔT in Clad, Average, °F	94	95
ΔT , fuel centerline to coolant, Average, °F	429	431

TABLE I
(continued)

	<u>Advanced Pulsed Reactor</u>	<u>Advanced Booster</u>
Fuel centerline temperature, average, °F	968	925
Average heat transfer coefficient, BTU/hr-ft ² -°F	22,500	22,500
Average heat flux on element surface, B/hr-ft ²	750,000	755,000
Temperature increase in fuel during pulse, °F	88	11
<u>Linac</u>		
Electron Energy, MeV		30
Average beam current during pulse, amps		33
Pulse duration, sec		3×10^{-6}
Source strength at 3×10^{15} neutrons/MW-sec, neutrons/sec		3×10^{18}

The core size for the pulsed reactor should be optimized by considering the opposing restraints of neutron lifetime and the capability to absorb the 1.1 MW-sec energy pulse. Until data is obtained on the fatigue life of fuel alloys with alternating stresses superimposed on a high steady state stress, it is impossible to develop an optimized core design. For a given size core and average power density, a trade off between increased repetition rate and burst energy can be made. The upper limit on the repetition rate is the speed of the rotating reflector wheel and is about 5000 rpm for 50 inch radius wheel. The lower limit on repetition rate for a fixed average power is imposed by thermal shock limitations. It is felt that the 1.1 MW-sec energy pulse selected for the reference design is near the maximum that could be attained in a 20 liter core. Thus if the thermal shock limit is lowered as more information

becomes available, dictating a reduction in the pulse energy, the repetition rate could be increased.

The characteristics of the Advanced Booster are determined by an estimate of the performance of a scaled-up standing wave linac such as the PHERMEX at Los Alamos. It is estimated that a charge of 10^{-4} coulombs could be accelerated with a reasonable scale-up. With the short filling time required, the machine could produce as much as 200 bursts per second. The characteristics of an assembly with a prompt multiplication of 250 driven by such an accelerator, with a 33 amp beam of 30 MeV electrons for 3 microseconds, are tabulated in Table I.

Since the amount of energy generated per pulse in the booster system is much smaller than that in the pulsed reactor, there is an opportunity to use a much higher repetition rate. The 6 liter core size selected for the Advanced Booster study may permit a repetition rate as high as 160 pulses per second which matches the capability of the accelerator fairly well. The prompt neutron lifetime in this core size results in a satisfactory pulse shape.

The neutron yield figures, summarized below, are based on an assumed value of 1.5 leakage neutrons per fission. This value is fairly typical of fast, high leakage, cores.

TABLE II
Neutron Yield

	<u>Advanced Pulsed Reactor</u>	<u>Advanced Booster</u>
Total leakage neutrons per pulse	5×10^{16}	2×10^{15}
Peak leakage neutrons per second	8.3×10^{20}	3.1×10^{20}
Integrated leakage neu- trons per second	1.1×10^{18}	3.2×10^{17}

3. DESIGN DESCRIPTION

The fuel elements in both systems are 3/16" O.D. pin type assembled into the core on triangular spacing with a pitch to diameter ratio of 1.1. Spacing is maintained with helically wrapped wires on each element.

Reactivity pulsing is accomplished by reflector sections carried past a bare face of the core by a rotating wheel. The pulsed reactor involves the use of two wheels. The primary wheel carries a thick reflector section and rotates at high speed to produce a sharp reactivity peak while the secondary wheel would carry a thinner reflector section past the core. The secondary wheel would rotate slower, at a speed dictated by the repetition rate, with the burst produced when both wheels are lined up with the core. The Advanced Booster would use a single wheel with multiple reflector sections attached around the periphery. The speed of this wheel need not be pushed to the limit since the pulse shape is determined primarily by the accelerator characteristics.

The performance of the booster concept is based on optimum location of the accelerator beam target at the center of the core. The average electron beam power of .480 MW implies a target volume, including coolant and structure, of about .3 liters.

Consideration of all the aspects governing core layout, such as electron beam entry and location of accelerator, fuel handling, rotating wheels, coolant inlet and outlet, and the location of neutron beam sources and experimental facilities, lead to some rather novel geometrical arrangements.

The liner between the stationary core and rotating device and the window for introduction of the electron beam in the booster are probably the most difficult design problems in the core with the exception of the fuel elements. These liners and windows should be protected from thermal shock effects and thermal gradients as much as possible. In the case of the liner between the core and moving reflector in both systems,

the gradient along the length of the window should be minimized. This requires a high coolant flow rate in this area to keep the coolant bulk temperature change small.

4. THERMAL AND HYDRAULIC ANALYSIS

A. Liquid Metal Cooled Core

The following limits were selected as the basis for the thermal and hydraulic analysis:

1. Power density = 1.5 MW/liter
2. Coolant velocity = 30 ft/sec
3. Pitch to diameter ratio of fuel elements = 1.1

The power density is in excess of that found in typical fast reactor systems; however, this may be feasible in view of the lower coolant temperatures involved in the pulsed systems. The pitch to diameter ratio of 1.1 is less than that used in most systems, but there is more of an incentive in the pulsed systems to minimize the coolant fraction in the core. This pitch to diameter ratio corresponds to 25 v/o coolant in the core (the minimum coolant volume fraction, with elements touching is about 9 v/o). Reduction of the ratio below 1.1 results in a severe decrease in the average film coefficient⁵. The heat transfer coefficients are based on data by Subbotin et al.⁶ obtained for a P/D of 1.1. Data below a P/D of 1.1 is sparse.

The fuel temperature gradients indicated in Table I are based on uniform power generation in the core and a fuel with a thermal conductivity of 10 BTU/hr-ft²-°F/ft, which is typical of uranium-zirconium alloys. The gradient would be reduced by a factor of about 8 for an aluminum alloy fuel and increased by about a factor of 1.7 for a UO₂-stainless steel cermet fuel.

In the Advanced Pulsed Reactor concept, the fuel can be relatively dilute. Cermets of uranium or plutonium carbide are under consideration. Metallic alloys of uranium with zirconium, aluminum, niobium, and palladium are also of interest. In some of these alloys, at 20 to 30 v/o fuel fraction,

it is possible to operate with all fuel in the gamma phase. At this lower fuel volume fraction, a larger core size is required for criticality. A paste type fuel of oxide or carbide particles mixed with sodium in fuel pins is also being studied.

The fuel in the Advanced Booster concept must be more concentrated because of the smaller core size and the fuel choice is limited to PuC, UC, metallic alloys of uranium, and paste.

The 400°F sodium coolant inlet temperature provides for a reasonable ΔT across primary and secondary loop heat exchangers while staying well above freezing temperatures in the secondary loop heat dump. An acceptable driving temperature gradient is also available for cold trapping in the primary system.

B. Rotating Fuel Package

As an alternate to the reference system, moving fuel sections are being considered. Thermal analysis⁷ of a gas cooled moving fuel package indicates that a power density of .15 MW/liter can be obtained with an atmospheric pressure gas coolant system. CO₂ is preferred over helium for this case because of its better heat transport properties. The fuel elements would be finned and the maximum fuel temperature would be about 1500°F in a UO₂-stainless steel cermet fuel. The gas would enter at 300°F and leave the core at 600°F.

Increasing the power density in the fuel would require pressurization of the gas coolant as well as higher fuel temperatures. Liquid metal cooling of the moving fuel would permit higher power densities; however, this is impractical due to the pressure generated at the periphery of the wheel if the coolant is internal and the erosive effects of the coolant if it impinges on the wheel from an external source. A scheme has been proposed wherein the liquid metal coolant is supplied to the moving fuel internally in the wheel and is discharged to the rotor cavity from the periphery of the wheel. The erosive effects of the discharged liquid and the problem of a sodium

vapor filled rotor cavity pose very difficult design problems.

Cooling of the rotating reflector wheel is not a severe problem in either system being studied.

5. MECHANICAL ANALYSIS

A. Rotating Devices

The rotating reflector wheels can be constructed with a fairly simple geometry. Stress analysis of a rotor with a radius of 50 inches using a maraging steel hub of hyperbolic cross section indicates that a speed of 5,000 rpm can be attained. The effects of neutron irradiation on maraging steel must be determined.

The analysis of a rotating fuel assembly is more complex. The wheel would have to be continuous around the periphery to provide for a coolant bypass leakage seal and must support the fuel bundle which would be operating at a high temperature. The problems in the fuel assembly involve centrifugal forces on the dense fuel as well as the thermal shock effects of the pulse. A rotating speed of 1,000 to 1,500 rpm may be feasible.

B. Thermal Shock and Stresses in Fuel Elements

The thermal stresses in the fuel alloy and cladding due to the average temperature gradient are calculated by classical techniques.

Evaluation of the effect of the thermal shock during a pulse⁸ depends on fuel geometry as well as fuel element manufacturing techniques. Two general approaches have been considered.

The reference designs have assumed a .005 inch sodium annulus between the fuel pin and cladding inside diameter. In this type of element, with a loose fitting clad, the fuel is assumed to expand independently of the clad. Thus the major stress in the clad involves the thermal stress due to the average temperature gradient through the clad with sudden temperature changes imposed on the inside surface by the heating of

the fuel during a pulse. The temperature damping effect of the sodium has been neglected and a constant temperature outside surface is assumed. The major stresses in the fuel pin are the thermal stress due to the average temperature gradient, which is tensile at the surface and compressive at the center, and the inertial stress due to the sudden temperature change during the pulse which alternates between compressive and tensile as the pin oscillates. In general terms, the inertial stress results from the inability of the fuel mass to expand as quickly as the temperature change dictates. The maximum value of the inertial stress is $E \propto \Delta T$ which is the compressive stress that would result if no expansion could occur during the temperature pulse. The reduction in stress below this maximum value is a function of the length and sonic velocity in a slender rod (or the fundamental period of vibration) in relation to the duration of the temperature pulse. Table III indicates the pin length required in a representative fuel alloy to reduce the inertial stress by a factor of 4 below its peak. It is assumed that the pin is supported at the center and the increase in temperature is linear for a period equal to the burst width at half power.

TABLE III

Length of Fuel Pin Required to Reduce Inertial Stresses by 1/4

	<u>ΔT Pulse, Seconds</u>	<u>Fundamental Period, Seconds</u>	<u>L, Inches</u>
Pulsed Reactor	60×10^{-6}	60×10^{-6}	2.4
Advanced Booster	6.5×10^{-6}	6.5×10^{-6}	.26

Restricting the length of the fuel pin to the indicated value might be feasible in the case of the pulsed reactor, but is difficult in the case of the booster. A summary of the fuel and clad stresses is given in Table IV for a loose fitting clad. Note that these stresses are based on average temperature gradients and would increase in direct proportion to the power peaking factor.

TABLE IV

Summary of Shock and Thermal Stresses in Fuel Elements

	<u>Advanced Pulsed Reactor</u>	<u>Advanced Booster</u>
<u>Clad</u> (stainless steel)		
Steady state thermal stress, psi	16,700	17,000
Cyclical thermal stress, psi	15,700	2,000
<u>Fuel Pin</u> (U-Zr alloy, E = 12 x 10 ⁶ psi)		
Steady state thermal stress, psi	25,000	25,000
Cyclical Inertial stress, psi	7,500/4	1,000

The mechanical properties used to calculate these stresses are not known accurately for irradiated material at the operating temperature and there is also a lack of high cycle fatigue data. In addition to the thermal shock stresses, the effect of the vibrating fuel pins on the integrity of the sodium bond must be considered. It is possible that gas within the element could be dispersed in the sodium eliminating the thermal bond at some location. Cavitation damage to the fuel alloy would also have to be considered.

The other approach to fuel element design involves the use of a tight fitting high strength clad to support the fuel material. With this technique there is less dependence on the strength of the fuel itself and cermet, ceramic, or paste fuels could be considered. High strength alloy clads such as A-286 seem to have the necessary strength. Stainless steel cladding material has proven satisfactory in the Fast Reactor Program in restraining the fission gas swelling of metallic fuels up to 8.7 a/o burnup. Paste fuels must be considered because of their desirable thermal and burnup characteristics. Dispersion of gas in the fuel and gross segregation or movement of fuel

within the pin due to vibration could cause severe problems. The thermal shock would develop a severe internal pressure shock loading on the container.

In view of the unique performance requirements for the fuel elements, there is a need for fatigue data on irradiated fuel and clad materials. In-pile testing would be required but its value is limited because of the impracticability of duplicating the service conditions. A combination of in-pile irradiations followed by hot cell mechanical testing will be required. Low cycle testing with thermal effects can be performed with electrical discharge heating while high cycle stress simulation is probably better done by ultrasonic techniques.

C. Sodium Technology

There do not appear to be any major problems in either system involving sodium technology. Corrosion effects should be relatively mild as a result of the low coolant temperatures. As the design evolves, however, the low temperature coolant might permit the use of materials with desirable properties that are not commonly used in fast reactor systems and in this case some data would have to be obtained. The process system components are reasonably sized and there should be no difficulty involved in constructing a reliable system.

6. PHYSICS

The critical masses have been determined from S_4 calculations⁹ and the prompt neutron lifetimes were obtained from diffusion theory. Various reflector materials were considered. Aluminum resulted in the highest critical mass and shortest neutron lifetime and was selected as the reflector material for this study. The effect of moderator beam sources in the reflector has not been considered but a thermal neutron absorber may be required to decouple the thermal neutron source from the fast core.

One dimensional analyses have been used to estimate the reactivity worth of moving fuel and reflector packages. Table

V lists the relative reactivity changes resulting from movement of a two inch thick iron reflector section and a two inch thick fuel assembly past opposite sides of the core in various combinations. The reactivity changes are listed relative to the first configuration.

TABLE V

Relative Reactivity Changes for Various Rotating Assemblies

<u>Reactor Configuration</u>	<u>Relative ΔK</u>
1. core-reflector	1
2. Reflector-core-reflector	1.93
3. core-fuel	1.34
4. Reflector-core-fuel	2.22
5. fuel-core-fuel	2.42

The moving fuel is assumed to have the same power density as the stationary fuel which poses a very difficult heat transfer problem. The reactivity effect of moving fuel at reduced power densities is being studied; however, it would seem that the benefits of using moving fuel are not great enough to justify the additional problems and expense involved.

For the pulsed reactor reference design a ΔK of .06 was assumed. This value can be obtained by using two reflector wheels. With a prompt neutron lifetime of 4×10^{-8} seconds a burst width at half power of 60 microseconds can be realized.

The pulse shape in the Advanced Booster system is more dependent on the accelerator source than the speed of the moving reflector. A neutron source yield of 3×10^{15} neutrons/sec per MW of electron beam power was assumed. The pulse width at half power is 6.5 microseconds.

The major engineering problems in the reactor physics area are concerned with burst reproducibility. Reliable instrumentation must be developed to indicate the magnitude of each burst and the system must be designed so that burst are reproducible with good mechanical control over the size of the burst.

7. CONCLUSIONS

The core designs for the Advanced Pulsed Reactor and Advanced Booster are based on current fast reactor technology and the results of the study indicate the unique aspects of the pulsed mode of operation.

In the pulsed reactor, there is strong incentive to operate at maximum power density and at maximum burst energy. The effect of both of these criteria combine to generate a complex cyclic stress and temperature pattern in the fuel element, and before an optimum combination of power density and burst energy can be determined, a considerable amount of information is required on the effects of this service on fuel and cladding materials. Within the limits of capability of the reactivity pulsing device, the burst energy can be reduced with a proportional increase in repetition rate to maintain a constant integrated neutron yield.

The same incentive towards high power density exists in the Advanced Booster; however, the accelerator limits the amount of energy per burst to a value well below the capability of the pulsed reactor. There is a vast improvement in pulse shape and the decreased burst energy can be made up to some extent by an increased repetition rate. The capability of the accelerator for highly repetitive pulsing and the less demanding requirements on the reactivity pulsing device give the Advanced Booster an inherent advantage in repetition rate over the pulsed reactor.

No direct comparison can be made between the experimental value of each system since this is so strongly related to the type of experiment that is of interest.

Some general design problems are discussed briefly; however, a rather detailed mechanical design would have to be accomplished before they could be defined with any precision. The most important of these are:

1. Design of core liners and windows
2. Cooling of experimental neutron beam source
3. Control and measurement of burst magnitude

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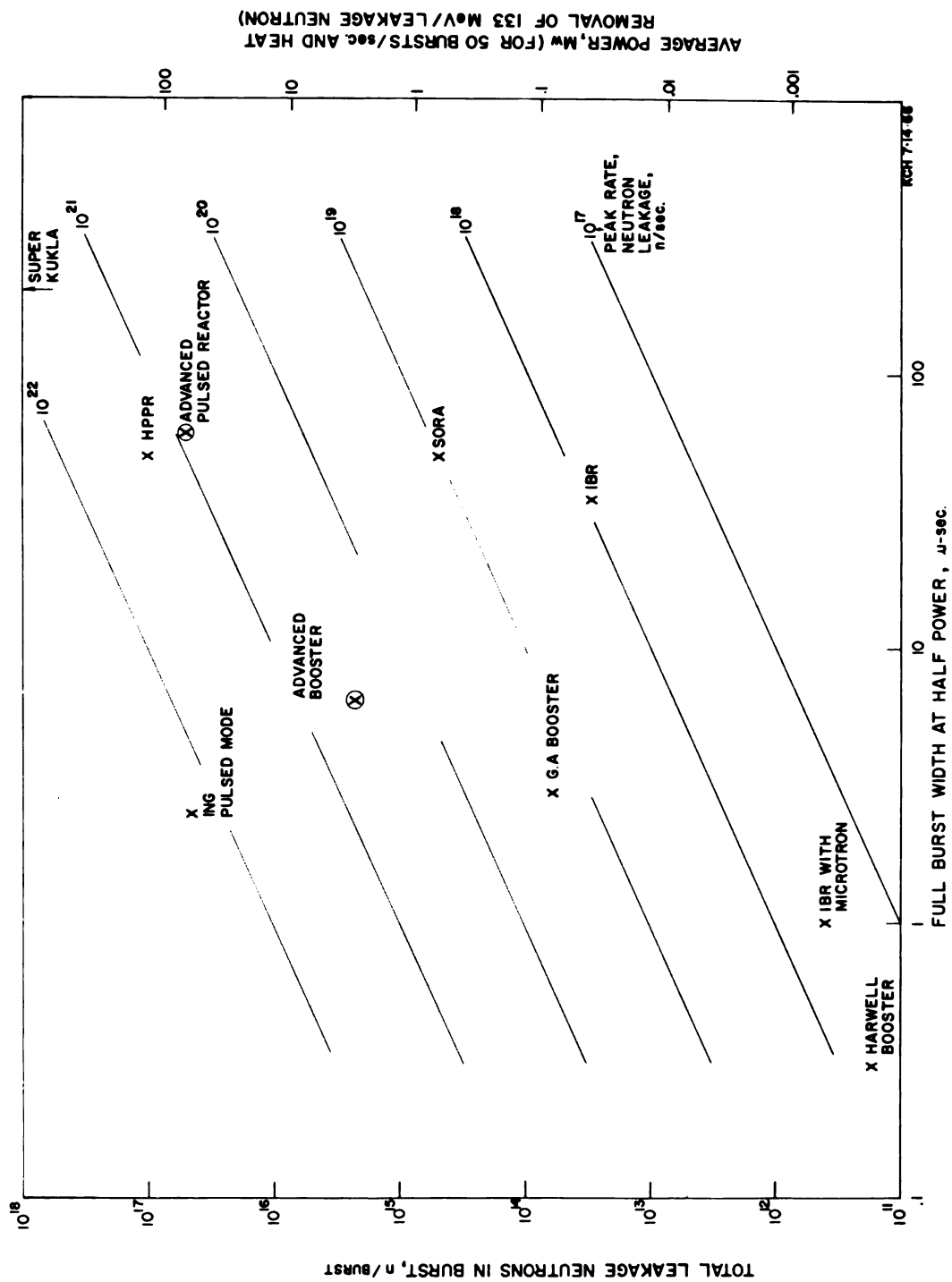
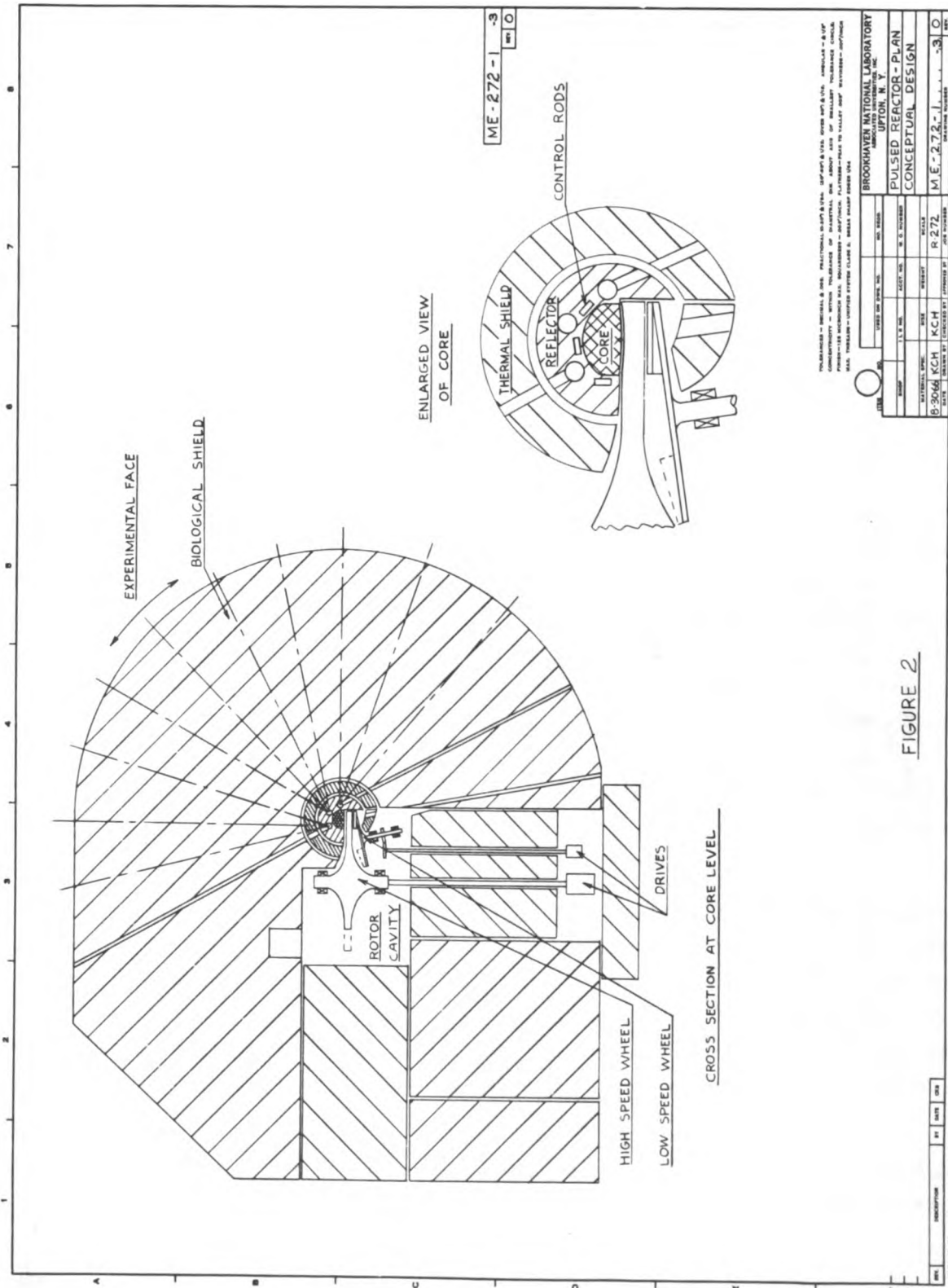
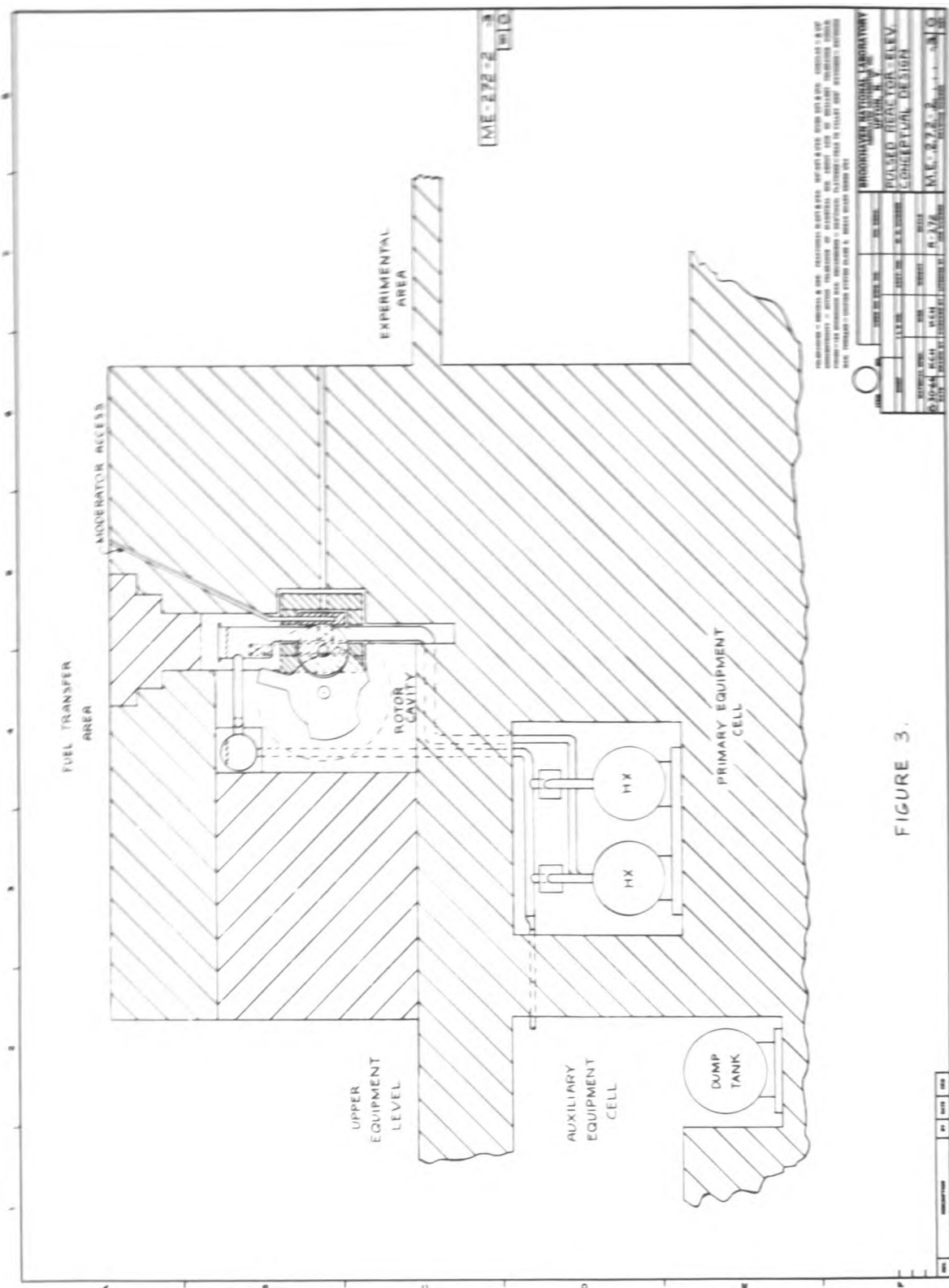


Figure 1. Characteristics of Various Pulsed Neutron Sources





DISCUSSION

OF

PAPERS III.B.1 (Raievski) and III.B.2 (Hoffman)

Chairman: H. Paxton

Secretary: T. Wimett

BURAS: We have had today several papers about repetitively pulsed devices having different repetition rates. I think it should be pointed out that, from the point of view of the solid state physicist who is interested in neutron diffraction by nuclear structure analysis, it does not really matter what the repetition frequency is. Let me make a very small calculation.

The IBR reactor working with a repetition rate of 5/sec has an average power of 6 kW. We can obtain there a neutron diffraction pattern of silicon in 2 1/2 hours with a flight path of 20 m and with a resolution of 1%. It is extremely easy to decrease this by a factor of five to a time of 30 min, or roughly 10^4 IBR pulses. In one pulse we have 5×10^{13} neutrons at the IBR reactor. Therefore if we had one pulse of 5×10^{17} neutrons, we would in principle obtain the same neutron diffraction pattern. With a more sophisticated approach, we could reduce this, I guess, to 10^{17} .

For conventional structure analysis it really does not matter if we take the picture of the neutron diffraction pattern in one burst or over 10,000 bursts. I am not going to propose a one-pulse device, but we would prefer to have as low a frequency as possible for other reasons which I mentioned earlier. I think that for some other experiments, such as nuclear physics or inelastic scattering of neutrons, this may not apply.

CONNOR: Being also a solid state physicist with some interest in the same techniques that Buras uses, I tend to disagree with him. What I should say is that, for a given duty ratio, the burst duration and repetition frequency are scaled together, but that a necessary additional scaling variable is detector size; with long bursts and lower repetition rates the detectors get very large. Within a reasonable range of extrapolation the cost of the detector actually becomes appreciable compared to the cost of the reactor.

RAMANNA: This question is a rather general one for anybody who can help me out and this concerns the use of storage rings for producing large currents of neutrons as a source for a booster.

BARTHOLOMEW: Well, we have looked very briefly into the question of storage rings in connection with the ING. Our conclusion was that if you were interested in repetition

rates of the order of 500 pps -- that is, dumping the storage ring at this rate with pulse widths of the order of $0.15 \mu\text{sec}$ -- it would be on the edge of possibility to build such a ring, that is, to find the phase space in a ring that could handle this current.

LARRIMORE: I would like to ask Hoffman if he has considered the use of liquid fuels or the paste or slurry fuels because I think a lot of people here are talking about it. We would like to know what your considerations are.

HOFFMAN: We did take a rough look at both liquid as well as paste fuels and, of course, these have the desirable characteristics of high coefficient of expansion that reactor physicists like in order to give them a good negative reactivity coefficient. Unfortunately, it is this high coefficient of expansion that results in very severe inertial forces that show up as a pressure at the bottom of whatever type of pin or vessel you want to contain your liquid fuel. Also, in the case of fuels like molten plutonium which are very concentrated and look good for booster systems, the density intensifies the problem in addition to these high acceleration forces. We took a look at paste involving a slurry of plutonium carbide or plutonium oxide particles 200μ in diameter together with sodium in fuel pins. Here again, the problem with the 100μ particles is that you have quite a large heat transfer area in the fuel, and the heat transfer to the sodium is so fast that these inertial stresses or inertial forces are built up again in the sodium. If one then goes to larger diameter particles in order to reduce this heat transfer area, one finds problems with individual particles or with the whole complex of particles stacked one upon the other. You get rather high contact stresses between particles that tend to cause particle attrition. Actually, there are solutions that one can postulate. For instance, if you do end up with this severe thermal transient in the sodium, you can postulate a stream of gas bubbling through the rod acting to absorb the thermal transient energy. This is a bit unpredictable because of the variation of fuel density throughout the core.

RAJEVSKI: I have a question for Hoffman regarding the rotor with horizontal axis. Have you gone far enough with the design of the window housing of the wheel? You must avoid the static pressure on the window and also the dynamic pressure which is generated at each pulse in this design. I get the impression that you have a very large flat window. How can you achieve the necessary rigidity?

HOFFMAN: It may not have showed up too well on the slide, but we did show a slight curvature to that window. It is concave inwards to the pressure of the sodium. This tends to be a three-dimensional problem in that the blade would have to be somewhat concave in that direction also and that there would be enough gap between the wheel and the window so as to avoid interference throughout the travel of the reflector. But, we have not gone very far in the design of the rotating device or of

the window. We want to get into the cooling problems involved in the rotating device so we can see just what gas pressure we need to get the heat out. Essentially, we are assuming that we see only the prompt gamma heating in the reflector section. We have to do quite a bit more detailed design on this problem. In one of my slides I showed you about six problems and I think we would like to spend two or three months on each of them.

FLUHARTY: The figures to be presented are data taken to investigate the thermalization time of neutrons in small moderators. The understanding of this process is considered to be of primary importance in the development of pulsed neutron sources. The people responsible for the success of the measurements are F. B. Simpson, Y. D. Harker, J. Mendzel, B. Malaviya and T. Gozani. The experiment used the RPI accelerator with the electron target in the center of the plane of the moderator slab. The target emitted fast neutrons which impinged on the polyethylene, and a chopper viewed the leakage neutrons perpendicular to the surface in the direction of the small dimension of the square slab. The chopper opened at different times after the "delta function" source neutrons were introduced from the target and sampled the neutron emission from the center or surface of the slab. The four samples studied were 10" by 10" with: 1) 3" thickness with neutrons from the surface; 2) 3" thickness with neutrons from a 1-1/2" re-entrant hole at the center; 3) 1-1/2" thickness with neutrons from the surface; and 4) 1-1/2" thickness with neutrons from a 3/4" re-entrant hole at the center.

Three dimensional plots of the data from the 3" thick slab with re-entrant hole are shown in Figures 1 and 2. In these figures the time-of-flight or wavelength is observed in the horizontal direction, while progressing towards the viewer is shown the phase of the chopper or observation time after the delta function neutron source. An observation time wavelength pulse ridge is seen as a function of observation time and wavelength. In Figure 2, it will be noted that the peak of the ridge moves to larger wavelengths till a time less than 40 μ sec it ceases to move to longer times and moves parallel to the observation time axis. The time at which this occurs describes the transition to equilibrium after which the population decays away with the fundamental mode of the system.

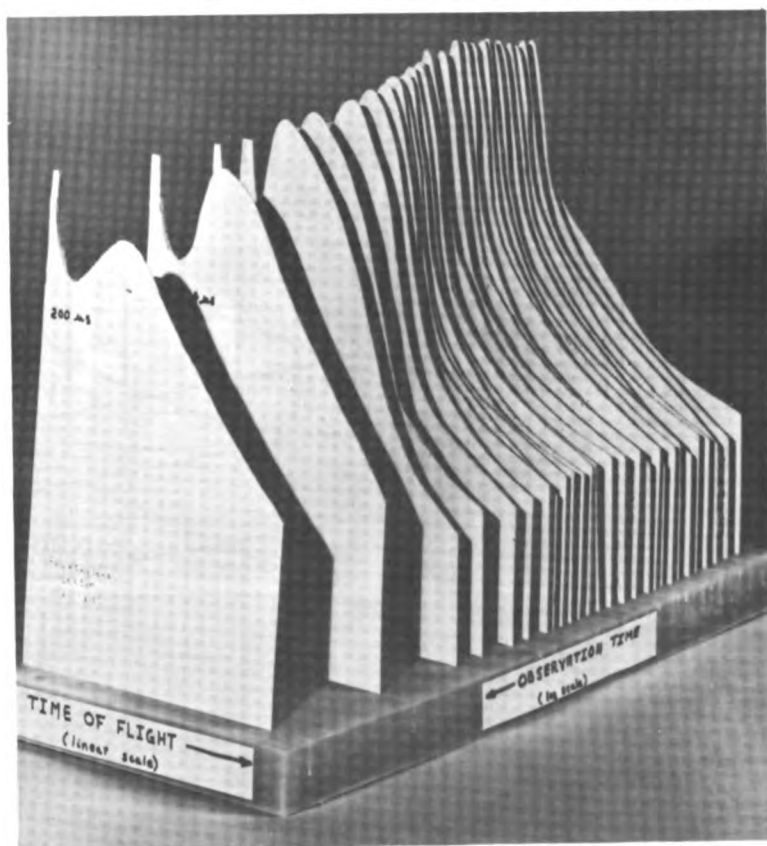
Selected data are presented in Figures 3 through 6 to show the time properties of these neutron wavelength pulses. The final exponential decays represent the fundamental modes decay from the systems. The peaks at the beginning of the distributions are the slowing-down-time distributions or, if you wish, moderation time functions. The preliminary character of the data is revealed by timing uncertainties but for the purpose of this discussion the width and amplitudes of these functions can be considered to be good. Disagreement as to the merits of accelerator pulsed sources vs mechanically pulsed systems revolve around the time-widths of the slowing-down-time pulses.

By comparing Figures 3 and 4 one notes that the surface intensity cannot be described by a single mode description. In this particular geometrical case, more

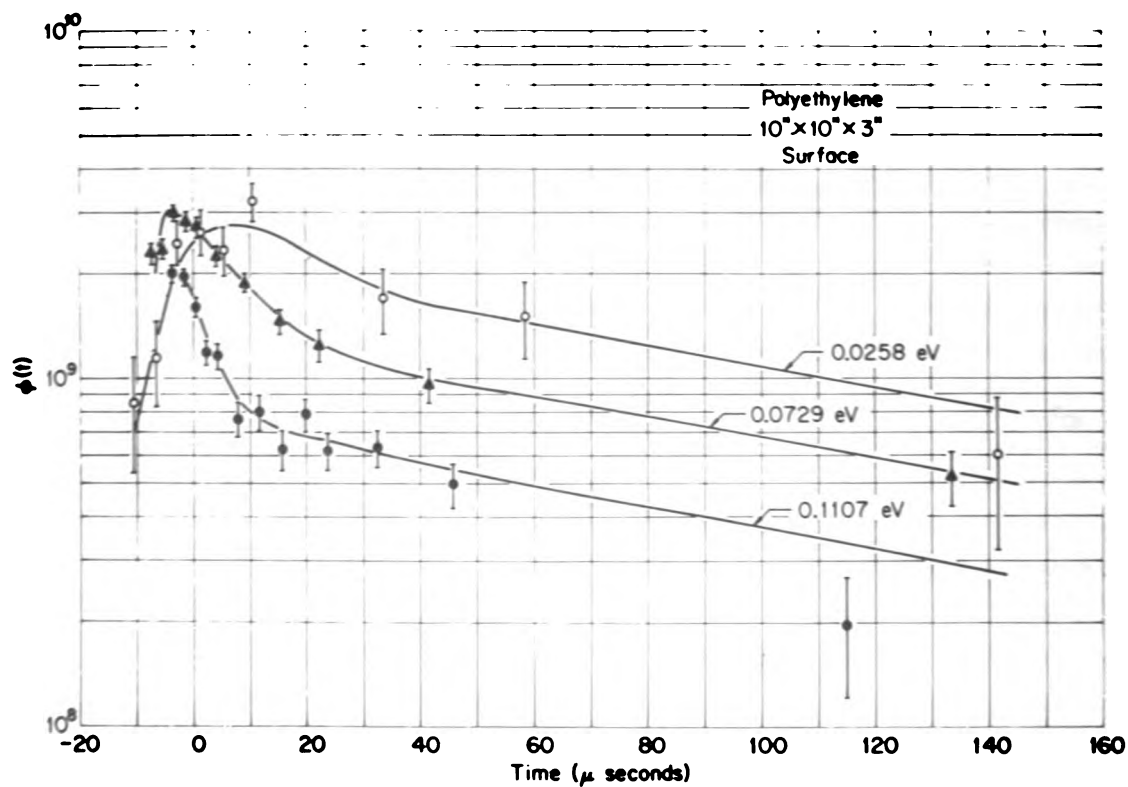
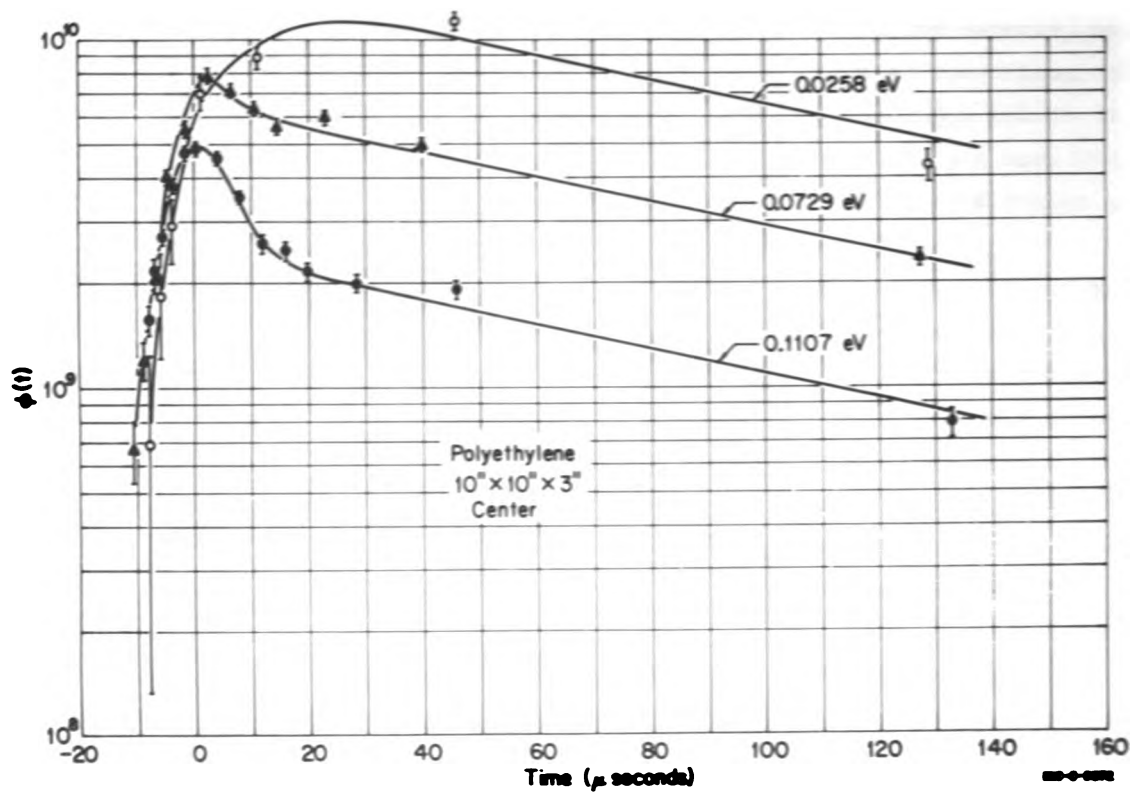
surface neutrons at higher energies than thermal are observed than one expects, and indeed the amplitudes of the neutrons from the surface are peaked in comparison with those from the center. The high leakage which is equivalent to adding a poison is readily seen in the fundamental mode decay. Comparing Figures 5 and 6, a more pronounced relative intensity is noted at higher energies for the surface vs center of the 1-1/2" samples.

In Figures 7 and 8 a comparison is made for 0.1-eV and 0.07-eV neutrons between the different sample measurements. This comparison shows a large neutron intensity from the surface of the 1-1/2" sample -- i.e., for 0.1-eV neutron there was almost an equal intensity to the 3" sample at the center! This must be explained as being due to the higher modal content of the system. Certainly for neutron energies down to thermal, advantages of short fast neutron driving pulses for resolution-intensity gains for time-of-flight experiments can be demonstrated. Figure 9 is a comparison for 0.025-eV neutrons between the different samples, and the advantage that appeared at higher energies has disappeared. The observed behavior is almost as an equilibrium spectrum except for the high leakage cases. A question that appears worth examining is that of the preservation of the slowing-down-pulse-width property of higher energies by lowering the temperature in the system.

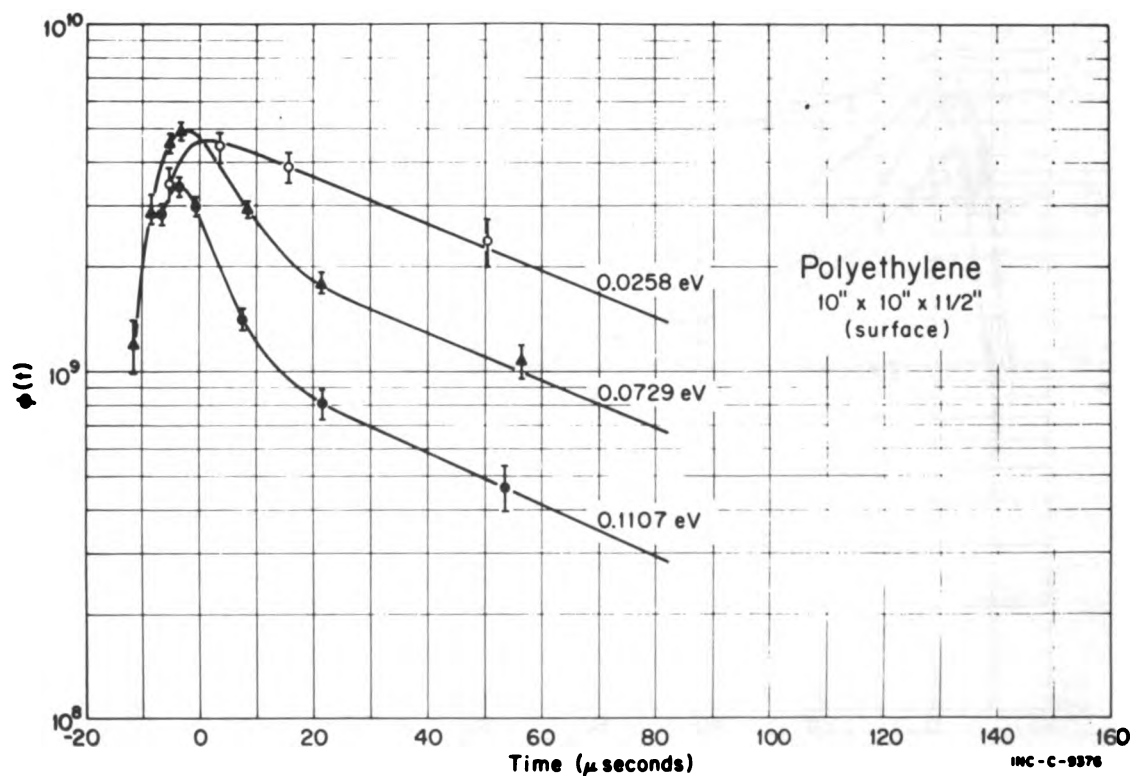
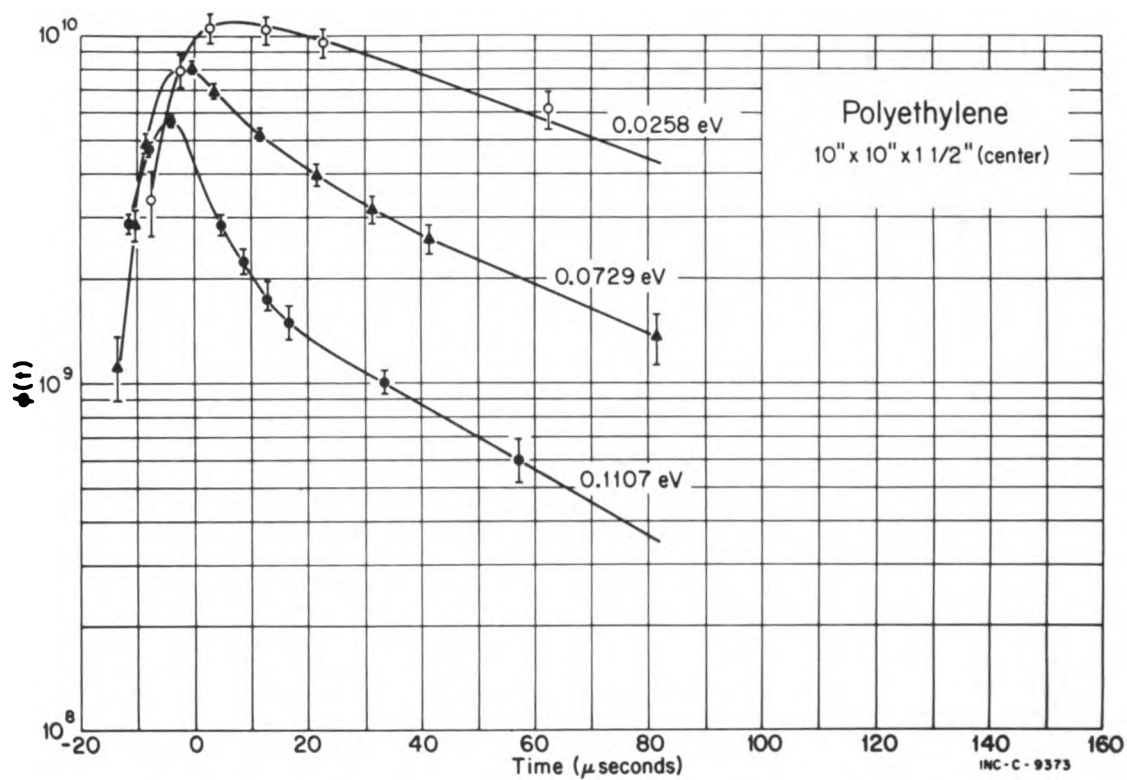
In Figure 10 some Russian data taken on water are shown. While it is not strictly proper to compare water with polyethylene, the important point is the differences in the amplitudes of the slowing down peaks compared to the equilibrium exponential decay. Also note the long time required to reach equilibrium. Compare these with Figures 3 and 6 for polyethylene with comparatively high leakage. These Russian data are for a large system of dimensions 12" by 12" by 15" where the behavior is viewed near the center of the system. The chopper resolution is 3.5 μ sec compared to 7 μ sec for the data presented here. This suggests that while the neutron slowing down process is similar in the two systems, the relative amplitude between 0.1 and the final steady state distribution is somewhat lower than for the high leakage system, suggesting a normalization problem. It appears that the slowing down times were short compared to the diffusion times for the neutrons to spread out and examine the edges of the large system. The initial slowing down distribution for the large system must be described by many higher modes, or possibly a thermal neutron shock wave exists until the fundamental mode is established. In the sense that the neutrons can be removed before these processes occur, significant gains for time-of-flight experiments appear to be possible through further studies.



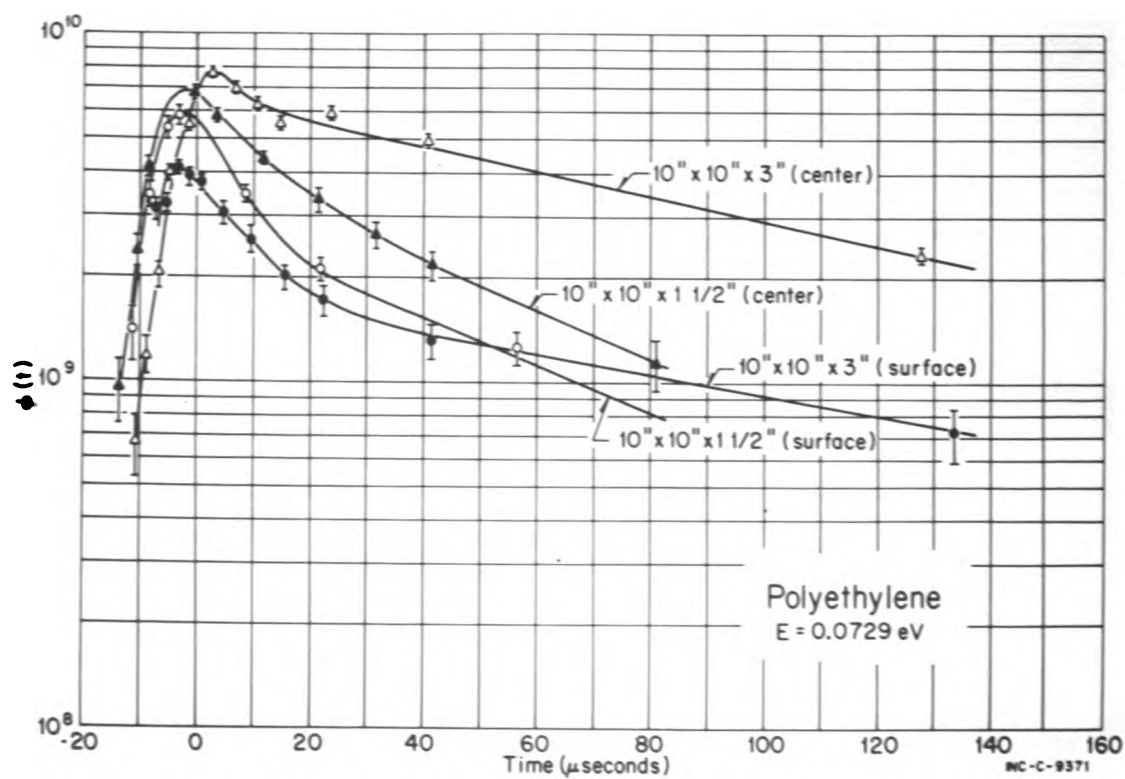
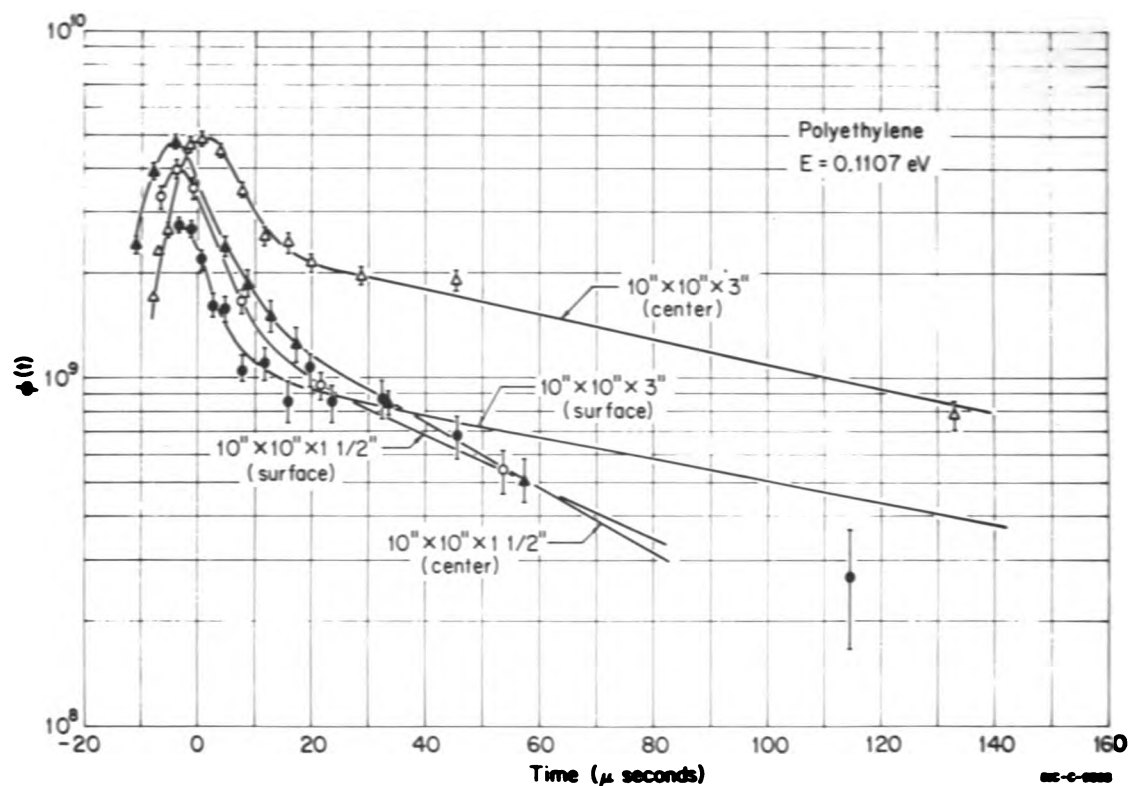
Figs. 1 and 2 Two views of neutron distributions from center of 3" x 10" x 10" CH₂.



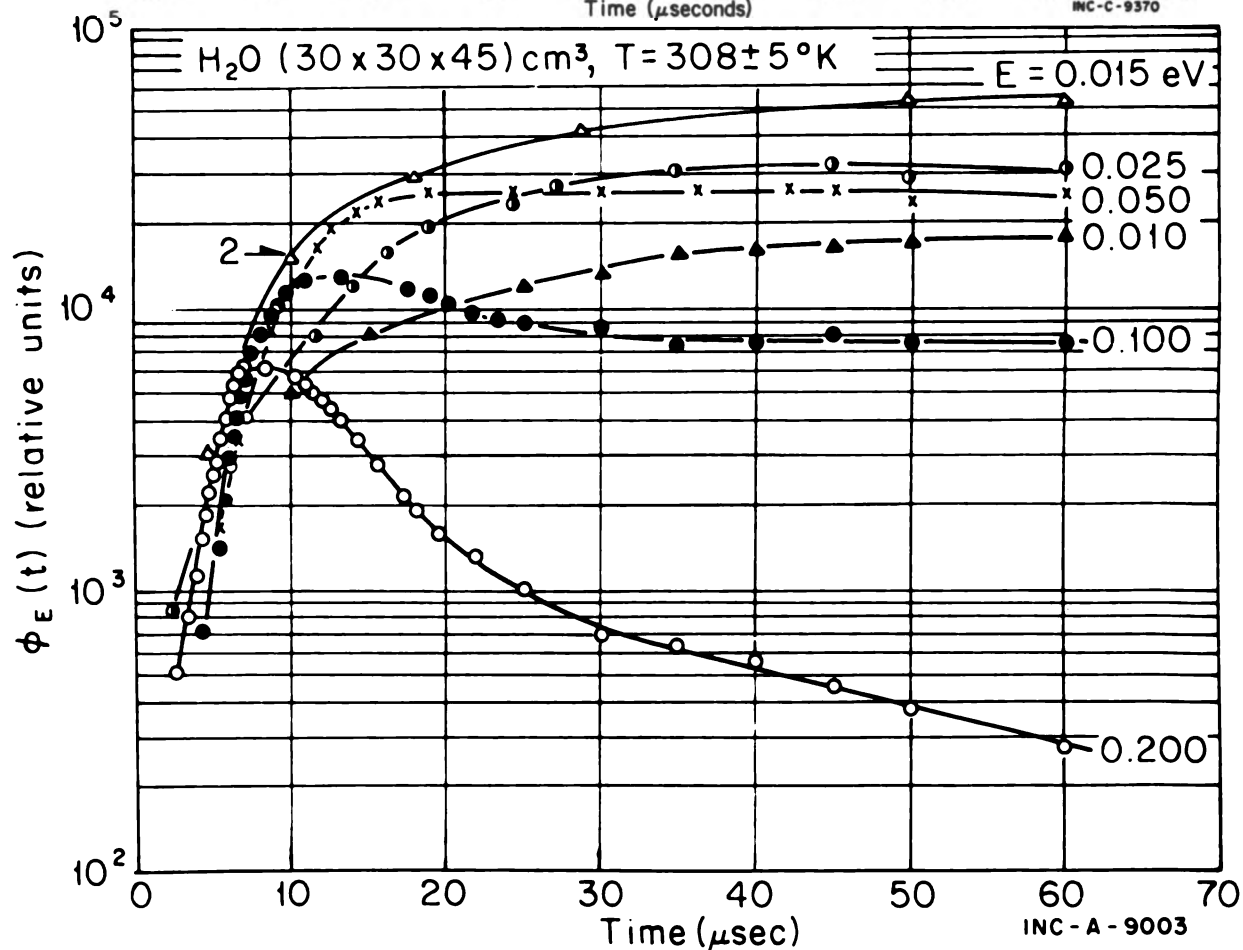
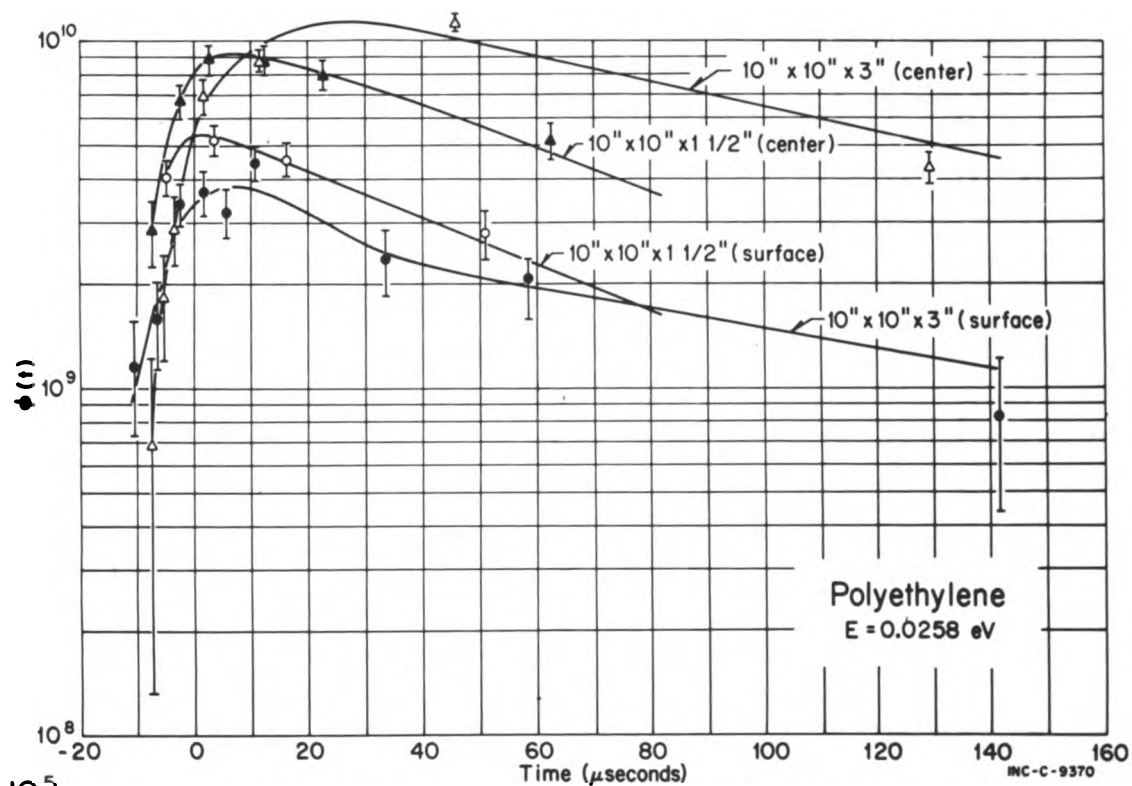
Figs. 3 and 4



Figs. 5 and 6



Figs. 7 and 8



Figs. 9 (top) and 10 (bottom)

THE NUCLEAR EXPLOSION AS A SINGLE BURST NEUTRON SOURCE*

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SUMMARY: Nuclear explosions have been used as pulsed sources for time-of-flight cross-section measurement. Because of high source intensity, good signal-to-background ratios have been obtained. The method is especially adapted to measurements with radioactive samples that produce undesirable background in conventional laboratory experiments.

1. INTRODUCTION

At a meeting assembled to discuss intense neutron sources it is appropriate to consider the most intense source that is known. A nuclear explosion can provide a mole of neutrons in a burst of 10^{-7} seconds duration and confined to a source volume of a few hundred cubic centimeters. Consideration of the uses of this source will serve to illustrate some of the ultimate applications of intense neutron sources. The source can be used in a very similar manner to other intense pulsed sources for time-of-flight neutron spectroscopy. The source duration of a hundred nanoseconds combined with a flight path of two hundred meters provides resolution of one half nanosecond per meter. This resolution is no better than that in use at several laboratories but the intensity is very much greater so that we can consider the special virtues of intense sources.

2. EXPERIMENTAL PROCEDURE

The experimental arrangement is illustrated in the first figure. The neutron source is a small nuclear explosive device buried a few hundred meters underground so that all radioactive materials will be contained safely below the surface of the earth. The neutrons are emitted in a short burst and a tightly collimated beam of neutrons is allowed to emerge through a series of baffles in a vacuum pipe and escape into an experimental area above ground. After the neutrons arrive at the ground surface but before radioactive gases escape, automatic valves close to seal all radioactive debris under the ground. When the neutrons arrive at the experimental area located in a tower at the ground surface, they are sorted out by their arrival times according to their energies. Neutrons that are all born in a few tenths microsecond with energies from a few electron volts to a few million electron

*Work performed under the auspices of the U. S. Atomic Energy Commission.

volts are spread out in arrival time at the ground surface into a time interval from a few microseconds to a few thousand microseconds.

The very great intensity of the source imposes conditions upon our work that are quite different from those in the usual laboratory experiments. If we are to obtain data from a single pulse from which we can plot a cross section versus energy with good precision over six decades of neutron energies, we must obtain many millions of events in a few milliseconds. It seems impossible to accomplish this task by counting individual nuclear reactions at the rate of thousands per microsecond. Instead we seek a detecting system in which an output current is proportional to the reaction rate of interest so that a plot of detector current versus time can be interpreted as cross section versus energy. We can illustrate the system by the measurement of a fission cross section as shown in Fig. 2. A well-collimated beam of neutrons a centimeter or two in diameter passes through a thin foil of fissile material. A few centimeters from the center of the beam a solid-state detector catches fission fragments from the sample. The average rate at which fragments strike the detector is typically hundreds of fragments in a tenth microsecond resolving time. The detector output is therefore a current that is proportional to the product of neutron flux and fission cross section. The current versus time is interpreted in terms of cross section versus neutron energy.

The neutron source is a small nuclear explosive with a yield in the range of 10^{23} to 10^{24} neutrons. The neutrons are emitted in a degraded fission spectrum with the possible addition of 14-MeV thermonuclear neutrons. Usually a moderator is used to enhance the flux of low-energy neutrons. The moderator design considerations are very similar to those in an accelerator experiment except for the heating of the moderator by radiations and shock from the source. If a moderator of polyethylene shielded with lead is placed 30 cm from the center of the source, it will typically be heated to a temperature of 25 electron volts and be given a velocity of 5 cm per μ sec. Under these conditions, the thermal neutrons are emitted from a rapidly moving hot moderator and the peak of the Maxwell distribution is at about 50 electron volts. The velocity of the moderator imposes a lower limit of 10 or 20 eV on the neutron energy that can be obtained in these experiments. After the neutrons leave the moderator they proceed up the evacuated flight path which has been designed to allow few wall-scattered neutrons to remain in the beam. Near the ground surface several small beams 1 to 10 cm^2 in area are formed by collimating apertures. These beams proceed in vacuum through the experimental area located in the tower placed at the ground surface.

Choice of sample size is dictated by beam and detector areas, each being a few square centimeters and the sample to detector distance may be 3 to 10 cm. The detectors are located outside of the beam to insure low backgrounds. Solid-state detectors are generally used for both fission fragments and gamma rays. Detectors

are chosen that are linear in response up to an ampere of output current and that can deliver a total of 100 microcoulombs of charge without degradation of the detector response.

Currents from the detectors are amplified logarithmically and transmitted down 300 meters of transmission line to recording stations. The signals are displayed on oscilloscope beams so that the beam deflection is proportional to the product of cross section and neutron flux. The beam is deflected only by the signal, no time base being used. An image of the moving oscilloscope beam is focused onto a photographic film which is moving rapidly at right angles to the line of motion of the image so that the moving film provides the time base for the photographically recorded signal. Figure 3 is a reproduction of a record showing signals from several detectors. Because of the logarithmic characteristic of the amplifiers, the excursions produced by resonances at late times correspond to two or more decades in signal amplitude. One oscilloscope beam on the photograph provides a base line reference and displays time marks for accurate determination of the time scale. The current scale for each detector is calibrated by imposing standard calibration pulses into the logarithmic amplifier input after all neutrons have gone by but before the shock arrives. These calibration pulses consist of nine signal amplitudes about a factor three apart and appear on the same film as the cross-section data. In this manner, the entire system, electrical and optical, is calibrated within a tenth second of the data recording.

The necessity of recording in the current mode imposes some severe limitations on our experiments. We cannot use coincidence techniques or pulse-height criteria to identify reactions or reduce backgrounds. However, we do have some very important advantages. The high neutron intensity makes sample-produced backgrounds unimportant. For instance, the spontaneous fission of a sample of ^{252}Cf produces a small background in a fission measurement. The neutron-induced fission rate will be much greater than the spontaneous fission rate. In a similar manner, thousands of curies of gamma rays from a tenth gram capture sample produce a negligible gamma-ray background compared to the gamma-ray production from neutron capture in a typical sample. The great neutron intensity also allows us to be extravagant in the design of the collimating and detecting system. The very small beam dimensions and small detectors give adequate detection rates with small "room" backgrounds from scattered neutrons.

3. DATA REDUCTION

The problem of determination of cross section versus energy from our analog signal of current versus time is similar to that involved in conventional time-of-flight experiments. Current and time coordinates of the signal are computed from data that have been determined from the photographic image and recorded on magnetic

tape. The (x,y) coordinates of the signal image on the film are recorded along with those of the reference base line, time marks, and the current calibration. A computer translates these coordinates into current versus time of the detector signal. At this point the data correspond to counts versus time channel of an accelerator time-of-flight data acquisition system and analysis proceeds in the same fashion.

4. NEUTRON FLUX DETERMINATION

The neutron flux must be known in order to determine partial cross sections. It is conveniently measured by reference to the ${}^6\text{Li}(n,\alpha)\text{T}$ cross section which is reasonably well known. A deposit of ${}^6\text{Li}$ on a thin foil in the beam provides a source of tritons and alpha particles that produces signals in detectors set up identically to the fission measurements. Ranges of the triton and alpha from this reaction are similar to fission fragments so it is convenient to use identical detectors in an identical geometry relative to beam and sample for both fission and flux monitoring signals. The flux can be determined absolutely from the measured currents if the amount of charge deposited in the detectors by the charged particles from the ${}^6\text{Li}(n,\alpha)\text{T}$ reaction is well known. This implies accurate energy loss calculations for ${}^6\text{Li}$ deposit and detector windows. Usual problems concerning the amount of ${}^6\text{Li}$ on the foil and the geometry must be handled. The flux versus neutron energy from the Petrel experiment is shown in Fig. 4.

In computation of a fission cross section from the fission signal and the neutron flux, we must also determine the charge deposited by fission fragments in the detectors including knowledge of the energy lost in the fissile deposit and the detector windows. An alternate method of determination of an unknown fission cross section which does not require accurate knowledge of energy loss in samples or detectors involves comparison of an unknown fission cross section to a known fission cross section such as ${}^{235}\text{U}$. In this method, deposits of the unknown and of ${}^{235}\text{U}$ are placed in the beam with identical foil thicknesses and detector geometries. The measured flux as determined by the ${}^6\text{Li}$ monitor is normalized to give the assumed ${}^{235}\text{U}$ cross section over the entire energy range and the ${}^6\text{Li}$ signal is now used as a flux monitor, whereas the reference cross section is ${}^{235}\text{U}$. In actual practice we have been able to reproduce the ${}^{235}\text{U}$ cross section within the accuracy of our measurement from a flux determined absolutely from ${}^6\text{Li}$ including the energy loss corrections.

5. RESULTS

Several well-known fission cross sections were measured on the Petrel event of June 1965.¹⁻⁵ These measurements were made as a proof-test of the method and were in general quite satisfactory. Cross sections were measured for ${}^{233}\text{U}$, ${}^{235}\text{U}$ and

$^{239,240,241}\text{Pu}$. The measurements extended from about 20 electron volts to a few million electron volts and appear to have about the same accuracy and resolution as the better existing fission data up to 20 keV. Above 20 keV, precisions of some laboratory measurements are much better than the Petrel data, but agree within the accuracy of our experiments. Figure 5 shows a sample of the Petrel ^{239}Pu data. The agreement with other measurements is good except in the deepest valleys where our relative freedom from background allows the determination of very small cross sections. The valley between 35 and 40 eV appears to reach a minimum of less than 0.06 barns. We measure 0.02 barns by counting the few individual fission events that occur during the 40 microseconds that correspond to the bottom of the valley and believe the cross section deduced from these events to be accurate within a factor 3. An indication of the freedom from background is given by the ratio of peak to valley of more than a thousand to one.

6. FUTURE USE OF THE METHOD

The main uses of the explosive neutron source will be in measurement of cross sections of very radioactive materials. We shall concentrate on measurements that cannot be made in the laboratory because of sample-produced backgrounds. Examples are measurement of fission cross sections of spontaneously fissionable nuclides and capture cross sections of intensely radioactive gamma-ray emitters. For example, measurements to be included in our next experiments are fission of ^{244}Cm and neutron capture cross sections in the resonance region for the fission product ^{147}Pm .

The incentive to develop the nuclear explosion techniques for use in time-of-flight experiments was the same as that for the other intense pulsed sources to be described here. We wanted to obtain a system with good energy resolution that would also have a favorable signal-to-background ratio even for radioactive materials. Our experiments so far have demonstrated the usefulness of the explosion source for cross section measurements in the energy region from a few eV to a few MeV. Signal-to-background ratios are very favorable, especially in the resonance region, and we are virtually free from sample-produced backgrounds.

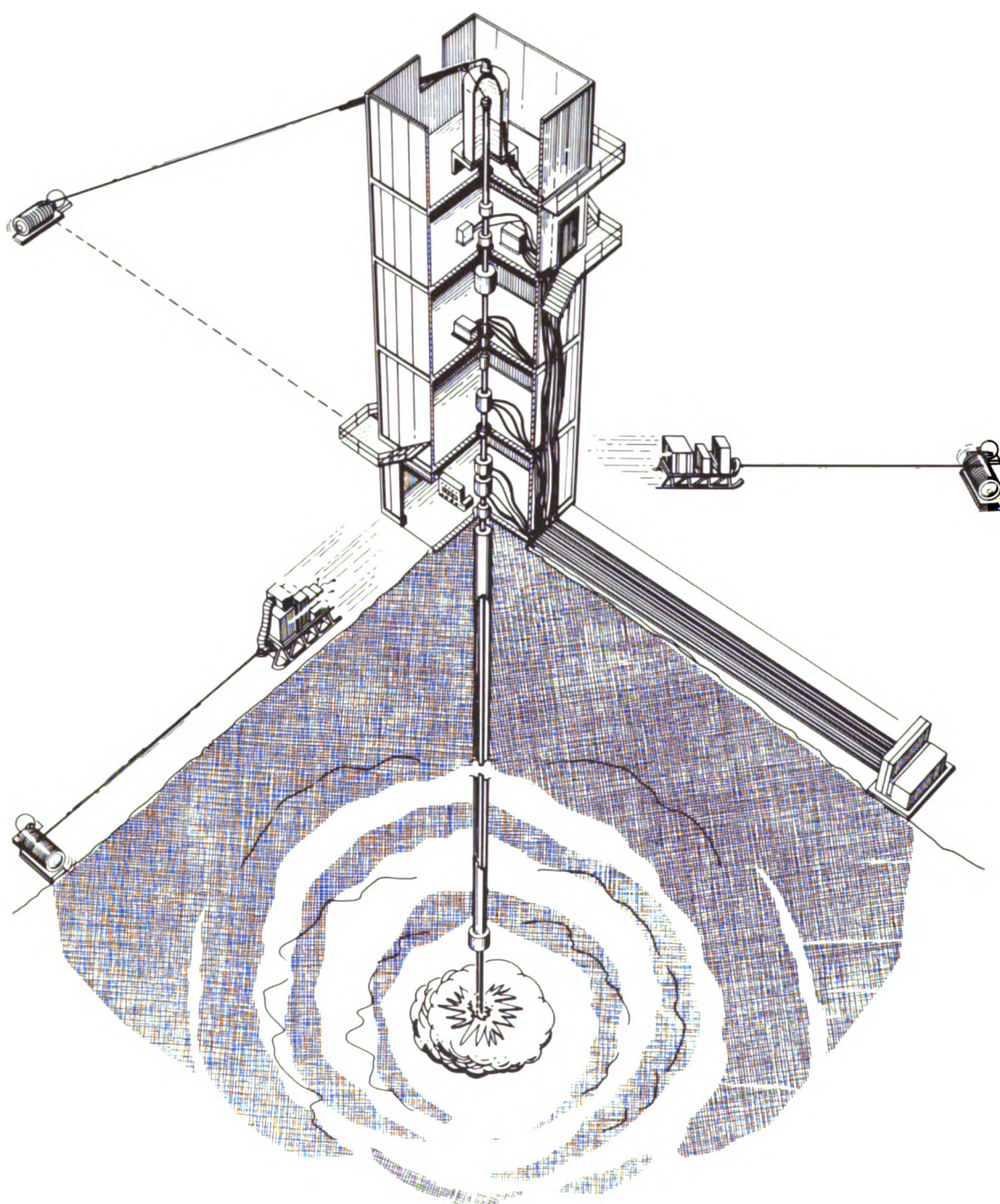


Fig. 1. The nuclear explosion source is located a few hundred meters underground below the experimental stations in a tower. Neutrons are released in a short burst by the explosion and travel up an evacuated flight path to the ground surface. Several small beams are formed by collimator apertures just below the ground and many measurements are made simultaneously by time-of-flight experiments located on the various floors of the tower. Valuable equipment and samples are pulled to safety on sleds after the experiments are completed, but before the ground collapses into a crater produced by the explosion.

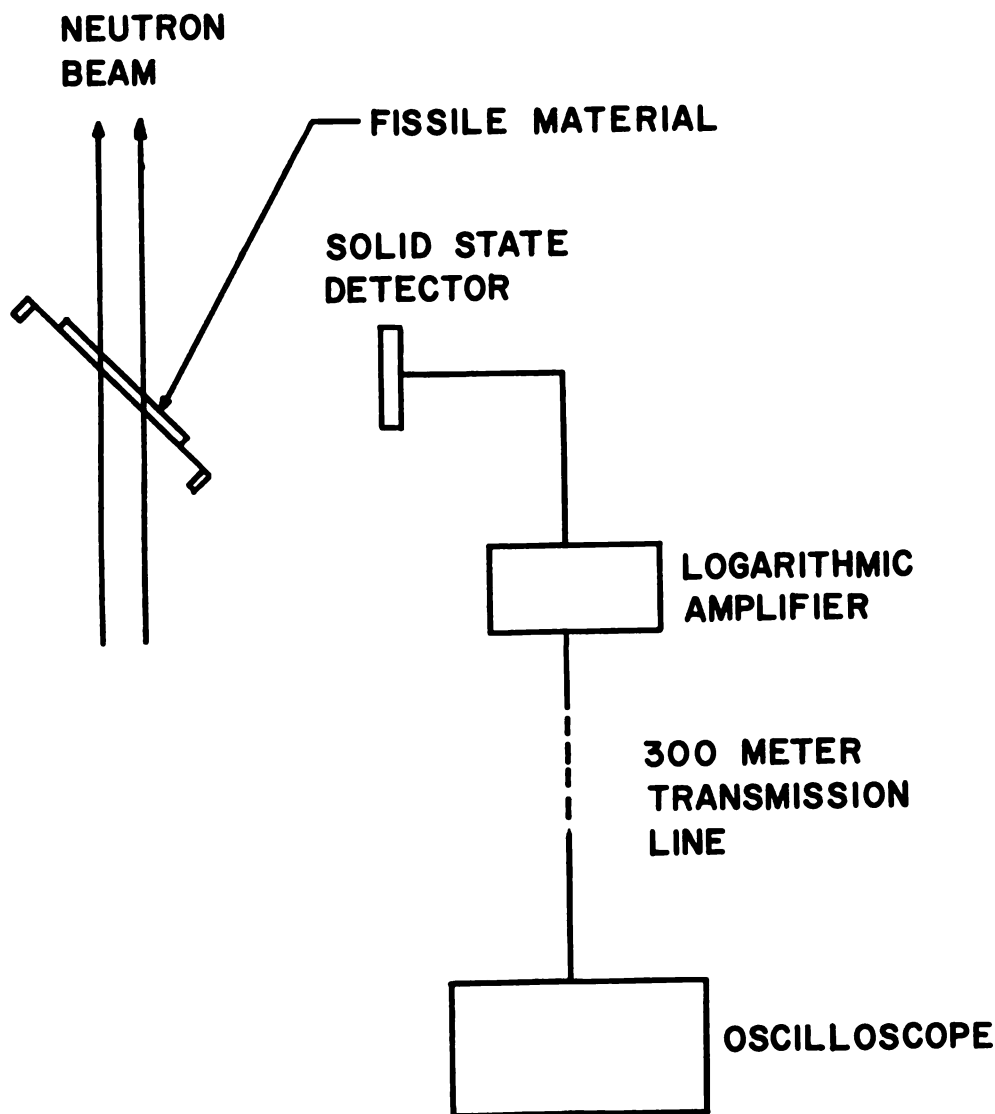


Fig. 2. Fission cross section measurement. A small, well-collimated beam of neutrons from the short explosive burst arrives at the experimental station with different energy neutrons spread out in time. The neutrons pass through the thin sample of fissile material and cause fission to occur. Fission fragments are caught by a detector placed well outside the neutron beam. The very large number of fragments per resolving time produce a current in the detector that lasts for the few milliseconds that neutrons arrive from the burst. The current is proportional to fission cross section times flux. After the signal is compressed by passage through a logarithmic amplifier, it is displayed on an oscilloscope and recorded photographically.

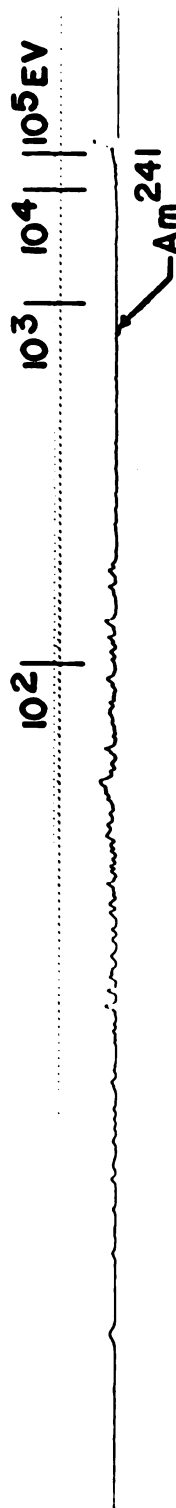


Fig. 3. Following logarithmic amplification, the signal is displayed as a deflection of an oscilloscope beam. No time axis is used on the oscilloscope so that as the signal fluctuates, it causes the beam spot to move up and down on a line. An image of the moving spot is focused on a photographic film moving at high speed perpendicular to the motion of the image. The motion of the film provides the time base so that the photographic record displays a plot of the logarithm of cross section times flux versus time. A second nearby oscilloscope beam provides a base line reference and time marks that are recorded on the same film. An energy scale has been drawn on the reproduction of a record of a fission signal from ^{241}Am . Time moves from right to left. Zero time occurs at the right, the vertical deflection being caused by fission neutrons. The sub-threshold fission appears in resonances in the center and left of the trace.

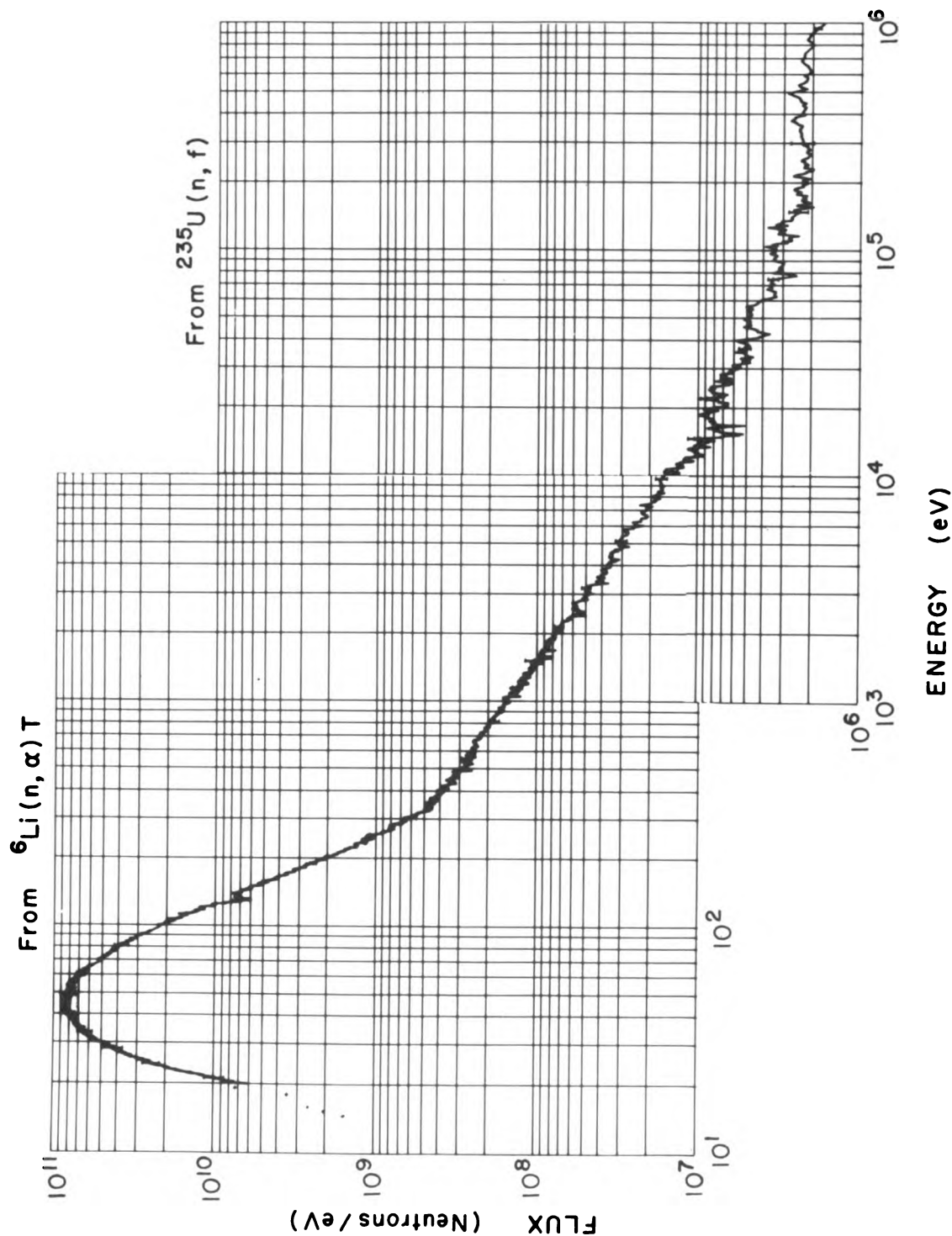


Fig. 4. Neutron flux in neutrons per eV in a three-square centimeter beam. The large group peaking at 50 eV are thermal neutrons from a hot moderator. From 300 eV to 10 keV is the 1/E leakage flux from the moderator and from 30 keV to 1 MeV the neutrons are fission spectrum neutrons that are transmitted through the moderator without scattering.

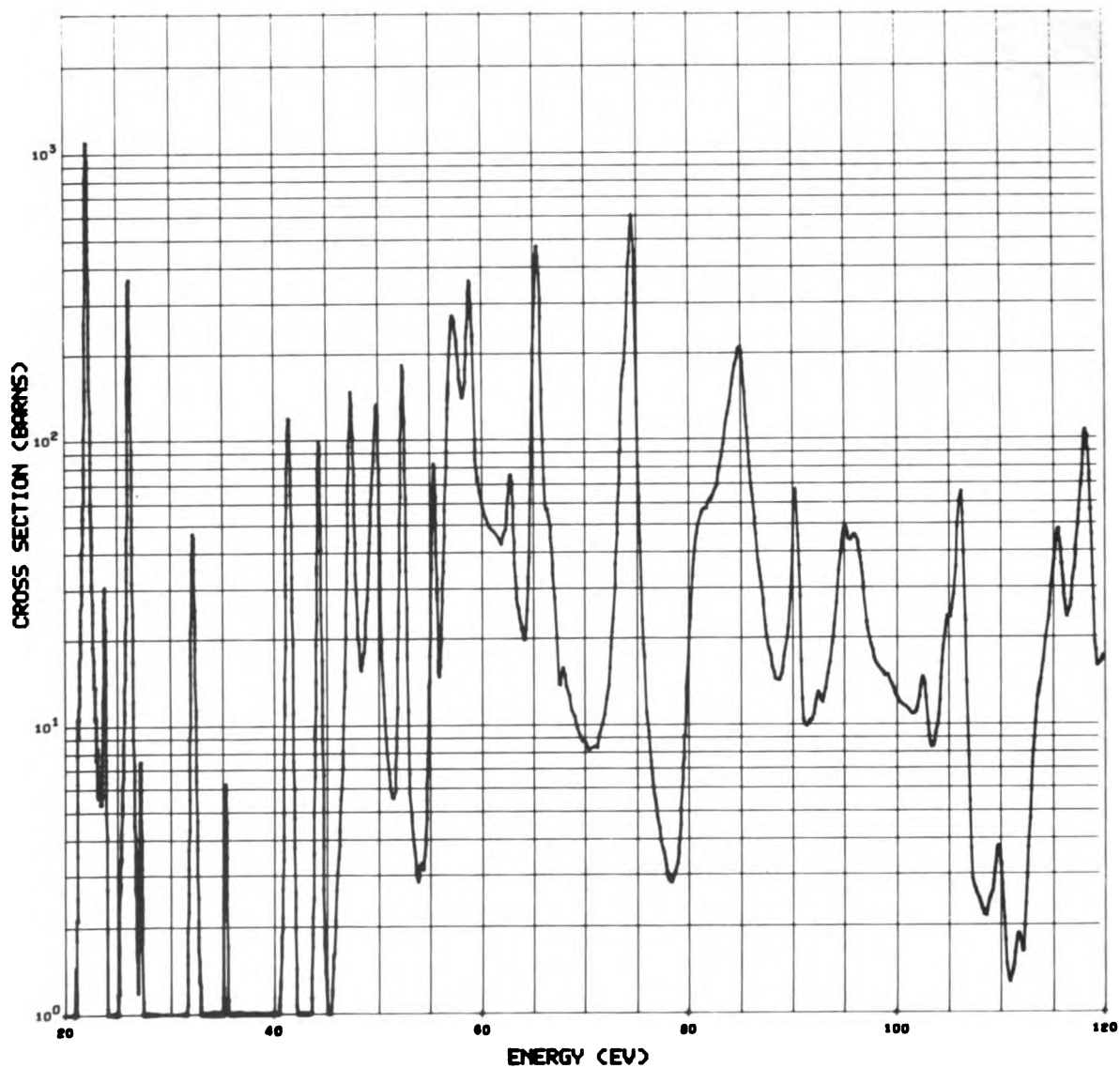


Fig. 5. Fission cross section versus energy for ^{239}Pu as measured on Petrel. Of special interest are the very low valleys which indicate the freedom from background. Between 35 and 40 eV the cross section dips below 0.1 barn.

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FEASIBILITY OF CERTAIN EXPERIMENTS
USING UNDERGROUND NUCLEAR EXPLOSIONS*

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SUMMARY: Two types of experiments are discussed; (1) Experiments with nuclei oriented at very low temperatures, and (2) direct measurements of fission lifetimes. In (1), it is concluded that such experiments, especially fission experiments, probably should not be performed using an underground nuclear explosion as a neutron source, because of unavoidable heating associated with the rapidity of the experiment; it is found that only a small number of unusual cases are at all feasible. In (2), a new experiment is proposed in which the passage of a recoiling compound nucleus through a crystal lattice may provide direct time-of-flight determinations of the lifetime against fission. The experiment utilizes the recently discovered "blocking" or "anti-channeling" effect in crystals. The feasibility of the experiment is discussed, with the conclusion that such neutron fission experiments are possible only with underground nuclear explosions as neutron sources.

(1) EXPERIMENTS WITH NUCLEI ORIENTED AT VERY LOW TEMPERATURES

Several experiments involving fission of nuclei oriented at low temperatures have been found to require very long counting times at reactors for their accomplishment(1). As a possible extension of these measurements, certain questions regarding the use of underground nuclear explosions have been investigated. Perhaps the primary difference between nuclear explosions and other methods of producing timed bursts of neutrons is the extreme rapidity of the entire process; regrettably, this feature appears to place almost insuperable obstacles in the path of most oriented fission experiments.

The basic problem is that during the few milliseconds following the explosion, enough heating is introduced by the fission fragments themselves to remove or seriously to reduce any nuclear alignment or polarization existing prior to the explosion. The lattice temperature of any reasonable target will rise to temperatures of $\sim 30^\circ\text{K}$ in a polarized neutron transmission experiment where essentially all fragments are stopped in the target. Only if the nuclear spin-lattice

relaxation time were at least one second at 30°K would such an experiment be feasible; this is a rather unlikely circumstance. For cases where, let us say, only 1 percent of the fragment energy is deposited in a thin target, no substantial advantage accrues. Since the lattice specific heat is, in almost all cases, a Debye specific heat which has the form

$$C = 464 T^3 / \Theta_D^3 \text{ cal/mol } ^\circ\text{K} \quad (1a)$$

the integral is of the form

$$Q = k(T_f^4 - T_i^4) \quad (1b)$$

and a reduction of Q by a factor 100, as indicated above, will only reduce the final temperature of the lattice in the present case to 10°K .

A reduction of the lattice heating by a factor of 10^8 would reduce the temperature rise to 0.3°K , which would probably be tolerable in most cases. This probably can be accomplished in cases of simple capture experiments where the nucleus deexcites only by γ -ray emission. This is also true for simple neutron scattering where no competing charged particle reactions occur. In all cases, rapidly acting shutters would be required to protect the samples from the initial burst of electromagnetic radiation from the explosion, and from undesired portions of the neutron spectrum.

In view of the difficulties inherent in this approach, it seems wiser at present to make use of the higher burst intensities which are now becoming available at electron linear accelerators, where such heating problems essentially do not occur, and to restrict the types of experiments to those which are feasible with intensities available at such machines.

The author gratefully acknowledges helpful conversations with B. C. Diven, L. Aamodt, P. Seeger, and others at Los Alamos, and with J. A. Harvey of ORNL.

(2) DIRECT MEASUREMENTS OF FISSION LIFETIMES

The possibility of the measurement of extremely short lifetimes in nuclear reactions by time-of-flight within single crystals has been suggested by Tulinov(2) and by Gemmel and Holland(3). The proposed method makes use of the so-called "blocking" or "anti-channeling" effect, in which a charged particle (in this case a reaction product) passes sufficiently near to a row or plane of atoms within the crystal to be deflected strongly from its original direction by Rutherford scatterings. A decay product from a nucleus which is recoiling (transversely) from its initial lattice position may be delayed in its emission sufficiently that its path, while initially in the same direction, may pass the members of the row or plane instead at a large enough distance to be "channeled" through the crystal, or at least be not deflected.

Experimental observations with emitted alpha particles(4), reaction-produced

protons(3), and elastically scattered protons(2,3,5) have given conclusive proof of the existence of the "blocking" effect, which typically appears as a decrease of more than 75 percent in intensity over a narrow angular range in the neighborhood of the direction of the row or more than 35 percent near a plane, perhaps with a small enhancement in the intensity at slightly larger angles. The total angular width β over which this "extinction" occurs may be estimated from considerations of screened coulomb scattering with various assumptions regarding lattice vibrations etc., and for most cases of interest yields values $\beta \lesssim 1^\circ$.

At the moment, there is still some uncertainty regarding the conditions under which "row" and "planar" blocking are important(2,5). Although both effects always occur, it appears that larger dips in intensity occur along rows (or axis direction in the crystal) whereas narrower patterns are observed in the various planar directions. In the latter case, pattern widths as small as 0.1° are found(3). We shall confine our attention to the planar case hereafter, because of intensity considerations.

We consider here a situation somewhat different from the previous suggestions(2,3) for observing very short lifetimes in nuclear reactions. If the recoiling compound nucleus moves through distances comparable to lattice spacings before emission of the reaction products, blocking by rows or channels adjacent to the one in which the compound nucleus originated can also occur. If the reaction product is emitted nearly along the planar direction by a compound nucleus which is recoiling in a transverse direction, a decrease in intensity may then occur when

$$v_r t \sin \theta \approx m l_t \quad (1b)$$

where v_r is the recoil velocity, t the time before emission, θ the recoil angle with respect to the planar direction, m an integer, and l_t the transverse spacing between planes.

As an example, consider neutron induced fission in single crystal $^{233}\text{UO}_2$ or $^{235}\text{UO}_2$ where the incident neutron direction is about 5-6 degrees away from a [111] axis; while the detector array is exactly aligned with a (perpendicular) (111) plane. We estimate or assume the following:

$$\begin{aligned} \beta_o &\approx 0.003 \text{ (} 0.2^\circ \text{)} && \text{oxygen plane} \\ \beta_U &\approx 0.007 \text{ (} 0.5^\circ \text{)} && \text{uranium plane} \\ v_r &= 5.9 \times 10^3 \sqrt{E_n} \text{ cm/sec (} E_n \text{ in eV)} \\ \tau &= \hbar / \Gamma_f \sim 6 \times 10^{-15} \text{ sec (for } ^{235}\text{U near thermal)} \\ l_t &= 1.65 \text{ \AA (to oxygen plane)} \\ &= 3.1 \text{ \AA (to uranium plane)} \end{aligned}$$

For the $^{235}\text{UO}_2$ case, one would then expect dips in intensity at E_n values given by

$$E_n \approx [m l_t / 5.9 \times 10^3 \tau]^2 \quad (2b)$$

$$\approx 220 m^2 \text{ keV.}$$

or at 220, 880, 1980 ... keV. Because of the structure of the crystal in this

example, these values represent alternately scatterings by oxygen planes and uranium planes; effects of heavy and light fragments would of course be indistinguishable.

Intensity considerations are approximately as follows. We assume a detector subtending a solid angle of roughly $2\beta_U \times 30\beta_U$, i.e. 0.5 cm x 30 cm at a 60 cm distance. The sample would be 0.7 cm x 6 cm x 10^{-4} cm thick, cut along the (100) planes, and mounted at $\sim 40^\circ$ so as to intercept a 3 cm^2 beam of neutrons from an underground nuclear explosion(6). For an assumed $\bar{\sigma}_f$ of 15 barns, the rate at which fragments reach the detector array would be approximately 5000 per microsecond, the statistical accuracy would be expected to be 3 percent for resolving times (0.1 μ sec) which give energy resolutions $\Delta E/E$ of about 0.01. These rates are near the limits of feasibility, since the count rates for oxygen planes are expected to show only a 40 percent decrease over a small portion (20 percent) of the detector area. For the uranium planes, a 40 percent drop over the central half of the detector may be expected, however. The above calculations strongly indicate that the only presently feasible neutron source for such an experiment is an underground nuclear explosion.

The detectors in the detector array could be made in a configuration of the same general type as the position sensitive semiconductor detector(7) but with a central strip electrode and two grounded edge strip electrodes on the back. The front electrode would contact the entire active area. The current induced by electron-hole pair formation within the detector would then divide via a transverse back resistance according to the distance of the fission event from the centerline of the detector, and yield a position signal from the ratio of the central electrode current to the total. In this way, a self-normalizing signal, with a relatively high degree of cancellation of statistical errors, should be available.

The major difficulty with the proposed experiment appears to be the spread in fission lifetimes. This can be expected to cause a serious broadening of the minima (in the detector current ratio) as a function of neutron energy. If this effect is sufficiently large to make the expected minima unobservable, resort may be had to the initial rise (with energy) in the ratio, which would occur at a neutron energy of roughly 1 keV for the lifetime assumed above. Here, a counting rate of 150 counts/microsecond would be expected, with a 3 percent accuracy in intensity and an energy resolution $\Delta E/E$ of about 0.03. This latter measurement would involve more complicated estimates of the flight path ($\sim 10^{-9}$ cm) and corresponds exactly to the previous suggestions(2,3).

If both effects could be observed, additional useful information could be obtained by intercomparison. Another possibility, especially in the case of ^{233}U , is that two different lifetimes, corresponding to the two compound spin states, might be seen.

The author gratefully acknowledges several helpful conversations with O. S. Oen, M. K. Robinson, and H. Lutz.

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 - (7) A silicon surface-barrier detector of this type is manufactured by Nuclear Diodes, Inc., Highland Park, Illinois, U.S.A.

ADDENDUM - added in proof

A detailed calculation (carried out on the CDC 1604) has been made for the total effect of all planes, both U and O (see UO_2 example above). Two cases were computed; the first assumed a single lifetime τ , the second assumed a χ^2 distribution of 4 degrees of freedom ($\nu = 4$) with an expectation value $\langle \tau \rangle$. In both cases summing over the several planes led to such a smooth variation in the signal that there appears to be no possibility of obtaining lifetime information by plane-to-plane time of flight. The results of these calculations are shown graphically in Fig. 1, where "signal" is plotted against $X = l_t / \langle \tau \rangle \cdot v_{\text{recoil}}$; l_t is designated as d on the graph. Of course, X is also proportional to the neutron time of flight, since the recoil velocity is proportional to the neutron velocity. The large dip to 0.7 in the special detector signal at low recoil velocities (large X) reflects "starting (U) plane blocking." It is seen that a substantially different shape of the curve occurs for the two assumptions regarding distribution of lifetimes. If the expectation value or average value of the lifetime varies only slowly with neutron energy, the shape of the curves can thus be used to make an estimate of ν for neutron energies of ~ 1 keV. At small values of X (high recoil velocities), the sum of the blocking by all planes, i.e., the sum of the small dips for the various planes, reduces the signal to ~ 0.98 whereas values much closer to unity apply when only the source plane and one other plane are included (upper curve). The lower (.98) asymptotic value gives a direct measure of the fraction of a long (many planes) path which is blocked from the detector. As such it can be used to determine the effective planar thickness over which blocking occurs. This value can then be applied to the width of the large dip in signal, and thus provides the needed scale factor for the lifetime determination.

Assumptions of the calculation were as follows:

Detector signal weighting function = $1 - |y|$, where y measured from centerline of detector.

U or O effective planar thickness = $\pm 0.05 d$ (lattice spacing).

Blocked intensity from U plane - 40% reduction over central half of detector;
weighted signal factor 0.3.

Blocked intensity from O plane - 40% reduction over central 20% of detector;
weighted signal factor 0.145.

The expressions used in the calculations were:

1) For single lifetime τ

$$S = 1 - 0.3 \left\{ 1 - e^{-.05x} + \sum_{n=2, \text{even}}^{\infty} \left[e^{-(n-.05)x} - e^{-(n+.05)x} \right] \right\} \\ - 0.145 \left\{ \sum_{n=1, \text{odd}} \left[e^{-(n-.05)x} - e^{-(n+.05)x} \right] \right\} . \quad (3)$$

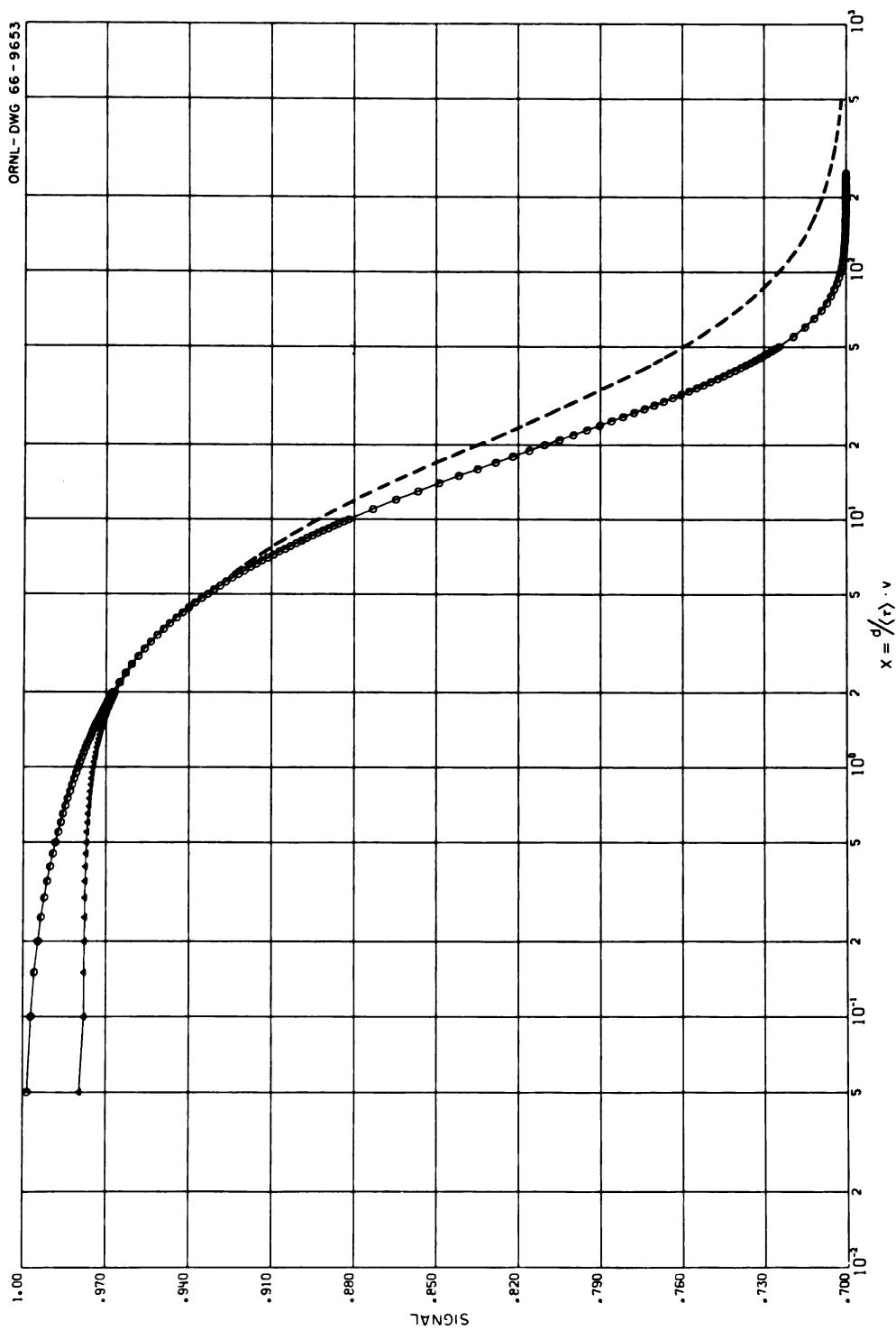
2) For χ^2 distribution $\nu = 4$, expectation value $\langle \tau \rangle = \hbar / \langle \Gamma_f \rangle$

$$S = 1 - 0.3 \left\{ 1 - \frac{1}{(1+.05 \frac{x}{2})^2} + \sum_{n=2, \text{even}} \frac{1}{(1+(n-.05)\frac{x}{2})^2} - \frac{1}{(1+(n+.05)\frac{x}{2})^2} \right\} - \\ - 0.145 \left\{ \sum_{n=1, \text{odd}} \frac{1}{(1+(n-.05)\frac{x}{2})^2} - \frac{1}{(1+(n+.05)\frac{x}{2})^2} \right\}. \quad (4)$$

A sharp cutoff for blocking was assumed; this is probably not a strongly unrealistic assumption. If channeling between planes is significant, the "weighted signal factors" above will be altered slightly.

It thus appears that three results could be obtained. These are effective blocking widths, average lifetime against fission near 1 keV, and the number of degrees of freedom in the distribution near 1 keV. The observability of two or more (disparate) lifetimes is not precluded by these calculations; a "stairstep" shaped main dip would be obtained in such a case.

The author is indebted to J. A. Harvey for a helpful discussion and to Mrs. S. Merriman for carrying out the computations.



DISCUSSION
OF
PAPERS III.C.1. (Diven) and III.C.2. (Dabbs)

Chairman: H. Paxton

Secretary: T. Wimett

KEEPIN: Using solid state detectors, in some cases don't you have appreciable neutron-induced, charged-particle reaction backgrounds, e.g., the threshold reactions $\text{Si}(n,p)$ or $\text{Si}(n,\alpha)$? Is your collimation really good enough to eliminate such background, or isn't this a problem in the energy range of interest?

DIVEN: There are few neutrons where the detectors are. The intensity of neutrons at the detector position is typically about at 1 part in 10^4 of the intensity in the beam. We do have some charged particle problems from our foil backing; (n,p) reactions in the backing on the foils have caused us some trouble at high energies. I should not have given the impression that we have no backgrounds at any energy. At early times, that is at high energies, we have quite significant backgrounds and some of them come from $\text{Si}(n,\alpha)$. We can see the effect of neutrons going through our detectors, but it happens that the (n,α) reaction does not amount to much compared to the scattering of neutrons off the silicon atoms which then give recoils that produce some current in our detectors. You would not see that in a laboratory experiment, because you could not see these little pulses. But we integrate them and record currents which appear as background. By the time we get somewhere below 10 keV, this background has disappeared except perhaps from some effects in foil backings.

BOLLINGER: Could you say something about how small a sample could be used in a total cross-section measurement? This presumably has something to do with the ability to collimate to a very small size.

DIVEN: We have used beams as small as 1 cm^2 . I have not really considered using very much smaller beams. The difficulty, I think, is that it takes one meter or so of material to collimate a beam well, and I am not sure whether or not we would run into a lot of trouble with wall scattering on the beam defining aperture. The design considerations for the collimating system are exactly the same as those in a laboratory experiment. I am afraid I could not say just how small a beam we would like to make.

BOLLINGER: Just to put my question and answer in the framework of what is done in laboratory experiments, it is perhaps worthwhile to remark that with a fast chopper there is no difficulty in working with a sample that has an area of roughly 1 mm^2 , so

that is the kind of thing one has to shoot for -- preferably a factor of 10 or lower than that if possible.

DIVEN: The same sort of area is perfectly reasonable here.

SAILOR: Do you have the means for measuring total cross section in the shots?

DIVEN: Yes, we did practice on measuring a total cross section on our Petrel shot and, I think, saw what the problems were rather clearly. We will not be able to measure a total cross section as precisely as it can be done in the laboratory because we can not do a sample-in, sample-out experiment using the same detector to see the signal with the sample in or out, so we have to measure the flux on the two sides of the sample with separate detectors to get the transmission, and we introduce an additional error there. When one measures a total cross section, what he hopes to deduce from this measurement is the resonance parameters. I believe that our method is especially suitable for measuring partial cross sections directly. I think we would be better off to measure directly the partial cross sections, say, fission, capture and scattering, to deduce the resonance parameters. We will, in the future, always measure the total cross section for just one sample on one of these beams because you have to perturb the beam with many samples to measure a total.

deBOISBLANC: Is that slide that you showed of an oscillographic trace recorded in time representative of the amplitudes that you get? It did not seem quite consistent that you could read that trace as accurately as your cross-section data implied.

DIVEN: It takes some doing. We try to read the position of the center of the trace to 10% of the line width. The line is perhaps not quite as bad as you saw there; these traces do not look as good in reproduction. We would like to determine all of our amplitudes on the film to about one part in 1,000 and, if we could do that, we would then have about 1% precision in determining the amplitude in the detector. That is because we have gone through a logarithmic system covering a very large dynamic range. We have not achieved 1% precision yet, and I do not think that we will ever have 1% accuracy. We might determine some currents to 1% but then there will be other difficulties that will limit us to, if I am optimistic, perhaps 5% accuracy in a cross-section measurement; if I am pessimistic, then 10%.

RAMANNA: Would you tell me whether you have carried out experiments to identify all the various types of particles that come out?

DIVEN: No. Of course, this method is in its infancy, and we have not done anything very elaborate yet. We have been content to take first those measurements that were easiest for us and try and learn how to do them. However, if we want to identify particles, I think we could do something about this. It is not as convenient as in the laboratory, because we can not look at individual particles and do a dE/dx and E

coincidence determination of the particle mass, for example. We probably can do some things of this nature using magnetic deflections.

BOLLINGER: As you pointed out, there is considerable difficulty in detecting complex events in a single shot of this sort. I wonder if you would respond in a somewhat more detailed fashion, going down the list of possible things one might like to detect and suggest possible ways of doing this or indicate if it is impossible.

DIVEN: Well, for example, you might want to observe the capture gamma-ray spectrum, from resonance to resonance. This immediately comes to my mind as a difficult experiment for us, but still possible. I think the intensities are just barely feasible. You could use a Compton spectrometer, but it would get to be extremely complicated if you wanted to have very fine gamma-energy intervals. Because each gamma-ray energy band would require its separate detector, or separate signal, and if you take more than forty or fifty cables and oscilloscopes and detectors, etc., it begins to be a rather complicated operation. To give you an idea of the amount of complexity that we can handle, for one of the four beams on one of these shots we would use forty or fifty detectors, cables, amplifiers and record data in two or three different time scales for each detector to provide seventy or eighty oscilloscope beams. But in doing this we have been measuring six or seven fission cross sections and a couple of capture cross sections all at once, and find some ten man-years in data analysis piled up.

BOLLINGER: How about measuring the mass distribution of fission fragments?

DIVEN: That is a nasty one with our particular method although of course George Cowan uses his radiochemical techniques and time-of-flight on a whirling wheel to do that kind of a job in the eV region. He spins a wheel of, say, plutonium in front of a thin slit and each resonance occurs at a different place on the wheel because the wheel is moving. He then saws up the wheel, dissolves separate pieces of wheel, and does his radiochemical analysis to find the mass distribution of fission fragments from resonance to resonance.

PERRY: You commented that your accuracy of determining the cross section would be 5 to 10% and I guess you meant at any given energy. Would you anticipate that the integral over restricted energy ranges would be much more accurate than that?

DIVEN: Possibly not. It will depend on what determines the accuracy, and since we are not quite that good, I can not say what our limitations will be for sure. I know at the present time one of our main limitations is in the calibrating system. I mentioned that we must read things with ridiculous precision in order to get a good answer, and this implies calibrating our entire system with very great precision. If it is an overall calibration that limits our accuracy, then we are hurt at all energies in the same way. If we take an integral over our ^{235}U fission cross section,

for example, that integral agrees with other people's integrals much better than the detailed areas under single resonances would agree.

deBOISBLANC: The thought of that piece of moderator filled up with neutrons fascinates me. Have you calculated roughly what the side leakage of low energy neutrons would be? Could the moderator flash past a hole in 1 μ sec? Might this not be a nice 1- μ sec burst of low energy neutrons?

DIVEN: I do not know. I have not looked into that. The moderators we have used have been effectively infinite in extent horizontally, about one meter across. This has to do with some hydrodynamic problems. We try to avoid punching a hole out of the moderator and squirting energy up the pipe which might cause some trouble. But that does not mean it could not be done. I had not thought of it because we deliberately attacked the problem in such a way that you can not look at the edge of the moderator. This moderator, if it were a cylinder with a diameter equal to its thickness, would expand more or less uniformly and very rapidly sideways as well as upward toward the experimenters.

ARMBRUSTER: I would like to ask Dabbs why is it not possible to determine the lifetime of the compound nucleus just from the width of the resonances? I think this a much more simple approach to the problem than this method that has just been proposed now.

DABBS: Of course, that is possible if one has sufficient resolution, but one must see all the resonances. I think this is an inherently interesting technique in that one sees the lifetime directly. In this experiment, of course, one always presumes that the lifetime varies smoothly with energy; if it does not, some very interesting jags and jumps might appear in this curve. If there are two disparate lifetimes, one might find two steps. There is some reason to think this is so, since Michaudon's recent measurements up to 250 eV on Pu^{239} rather clearly indicate this. I might mention that it is possible also to do this type of experiment using (d,p) fission; I do not think it is appropriate to discuss that case at this meeting, but one does have an additional parameter there; you can vary the compound nucleus excitation and the recoil velocity independently.

RAMANNA: I presume that by this method you get the lifetime for fission only for the case of fast neutron fission.

DABBS: Yes, the restriction is even more severe than that. You can only do it in the neighborhood of an energy which is related to the lifetime, that is, there is a correlation, and for a lifetime similar to those observed in thermal fission that is around 1 kV. So the method is really rather restricted.

SESSION IV

NON-FISSION DEVICES FOR THE PRODUCTION OF
HIGH INTENSITY NEUTRON BURSTS

CYCLOTRONS, SYNCHROCYCLOTRONS, AND SYNCHROTRONS
AS PULSED HIGH INTENSITY NEUTRON SOURCES

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SUMMARY: Cyclotrons, synchrocyclotrons, and synchrotrons are not extensively used as pulsed neutron sources because their inherently pulsed nature is a disadvantage for most experiments. The success of the mechanisms used to remove the beam microstructure and the spreading of the beam because of phase stability in the synchrotron makes existing synchrotrons not very useful as pulsed neutron sources. The use of high energy protons to produce neutrons increases the length of the source because the range of the proton increases rapidly with energy. Examination of the geometry of neutron spectrometer collimating systems and source and detector sizes leads to the conclusion that the maximum energy protons useful for neutron time-of-flight spectroscopy is about 1 BeV. No single criterion appears suitable for comparing cyclic particle accelerators as pulsed neutron sources. The time-of-flight field has been divided into slow and fast neutron time-of-flight spectroscopy on the basis of the neutron energy response of the detector. The slow neutron detector responds to thermal neutrons and the fast neutron detector has a neutron energy threshold. The criteria finally chosen for comparisons are the maximum intensity of the burst and the average neutron intensity when the accelerator is used as a neutron spectrometer. The accelerators now used for neutron spectroscopy, proposed improved existing accelerators, and proposed accelerators are compared as neutron spectrometers.

1. INTRODUCTION

Cyclotrons, synchrocyclotrons and synchrotrons are not used extensively as pulsed neutron sources, principally because the scientists who are in charge of operating these accelerators are interested in using them for other experiments. However, whenever one of these magnetic resonance accelerators is used as a pulsed neutron source, it has proven a very intense and useful source. This fact is not surprising because all cyclic particle accelerators are inherently pulsed devices and, when one takes advantage of one of the natural characteristics of

a system, the system is usually being used most efficiently.

The inherently pulsed nature of cyclic particle accelerators is usually regarded as a great disadvantage by the experimentalists who use them and great care is taken to remove the sharp pulses and obtain a more uniform beam as a function of time. In fact, in the considerations of the Bethe Committee on Meson Factories, a great deal of attention was given to the duty-cycles of the various accelerators proposed. The reports describing the proposed accelerators devote several pages to show how the microstructure and macrostructure of the pulses can be smoothed out so as to increase the duty-cycle of the accelerator. In describing the Brookhaven Alternate Gradient Synchrotron, Dr. M. H. Blewett¹ states, "During the acceleration, the beam is very tightly bunched into pulses of 15 to 20 nanoseconds duration." She later explains that "the bunched structure can be removed by programming an exponential decay in radio frequency amplitude at the end of the acceleration period and allowing the beam to drift inward to hit the target. By this method, bursts of 10 to 15 milliseconds duration can be obtained."

Because of the elaborate and successful schemes used to remove the pulsed characteristics of synchrotrons, none of the synchrotrons which accelerate particles to energies greater than 1 BeV would be very useful as intense pulsed neutron sources. In addition, geometrical considerations of source length, which I will later discuss, show that accelerators with energies greater than 1 BeV are not very useful for neutron time-of-flight spectrometry. However, cyclic particle accelerators of lower energies have been very useful in neutron time-of-flight spectroscopy, even though the accelerator designers have attempted to "design out" the characteristics which are useful for this work. If the time-of-flight work were taken into consideration in the design of a cyclic particle accelerator, it could be a very much more potent tool in this field than it is.

2. THE TARGET AND NEUTRON SOURCE

In the last few years the production of neutrons by high energy protons has received increased attention because of the demand for greater flux needed to study the effects of very intense neutron radiations on the components of nuclear energy systems. The ultimate limitation on the production of high fluxes with reactors appears to be the rate at which energy can be removed from the volume where the neutrons are produced. Drs. Bartholomew, et al., at Chalk River, have pointed out that the energy which is deposited in the target to create a neutron by using a 1,000 MeV proton to bombard a bismuth target is about 30 MeV, rather than the approximately 100 MeV which must be deposited in the target to create a neutron by the fission process.² The use of high

energy protons for the production of a high intensity pulsed neutron source is not new,^{3,4} but their use as a continuous neutron source as a substitute for a very high flux reactor is new and the first results published using this process has led to the recent interest in this subject. Data about the production of neutrons by high energy protons has improved substantially in the last year through the experiments of Gibson, Zucker and Gross of the Oak Ridge National Laboratory and Green, Frazer, Hillburn and Milton of the Chalk River Nuclear Laboratories.

The number of neutrons produced per incident proton as a function of the proton energy in heavy targets is shown in Figure 1. The energy required to liberate a low energy neutron by the bombardment of a lead target with high energy protons is shown in Figure 2. These data were taken from an Oak Ridge National Laboratory progress report.⁵ These results show that the total number of neutrons produced in a thick target increases linearly with the energy of the incident proton. From this information alone one would conclude that if an extremely high intensity neutron burst was desired, then the energy of the particles incident on the target should be as high as possible. Then why not use the 33 BeV AGS as a very intense neutron source? Unfortunately, as the energy of the proton beam increases, other factors limit the usefulness of the neutron source.

Neutron physicists are used to thinking of the sources produced by accelerators as point sources. If one assumes that the neutron source is a point source, the higher the intensity of the source, the better it is. What really counts, however, is not the total number of neutrons produced by the source, but the number of neutrons which are counted in the detector. For a point neutron source, if we assume that the neutrons are emitted with spherical symmetry from the target, the intensity at the detector is limited either by the width of the slit of the collimating system or by the size of the detector. However, for an extended neutron source, one has to look more carefully at the geometry of the system in order to determine the maximum effective source length. The idealized geometry for three different configurations is shown in Figure 3. The point source geometry is shown in 3a, the geometry of a system which is not limited by the size of the detector is shown in 3b, and the more usual geometry where the detector size limits the detectable intensity is shown in 3c. I have used R for the length of the target to indicate the range of the particle in the detector, s for the width of the defining slit, d for the length of the detector, l_{sd} for the distance between the source and the

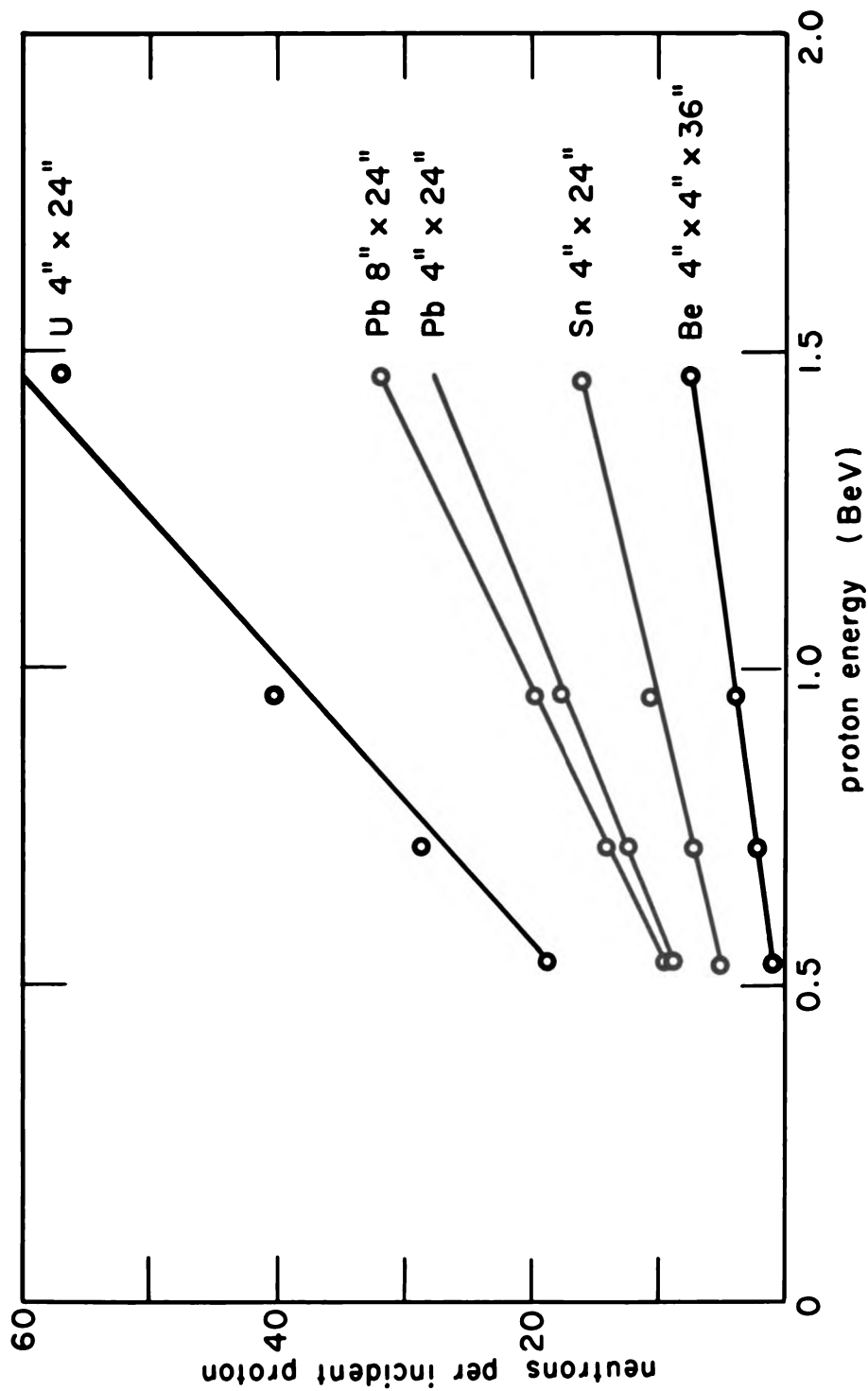


Figure 1. The number of neutrons produced per incident proton from bombardment of thick targets as a function of the energy of the incident proton.

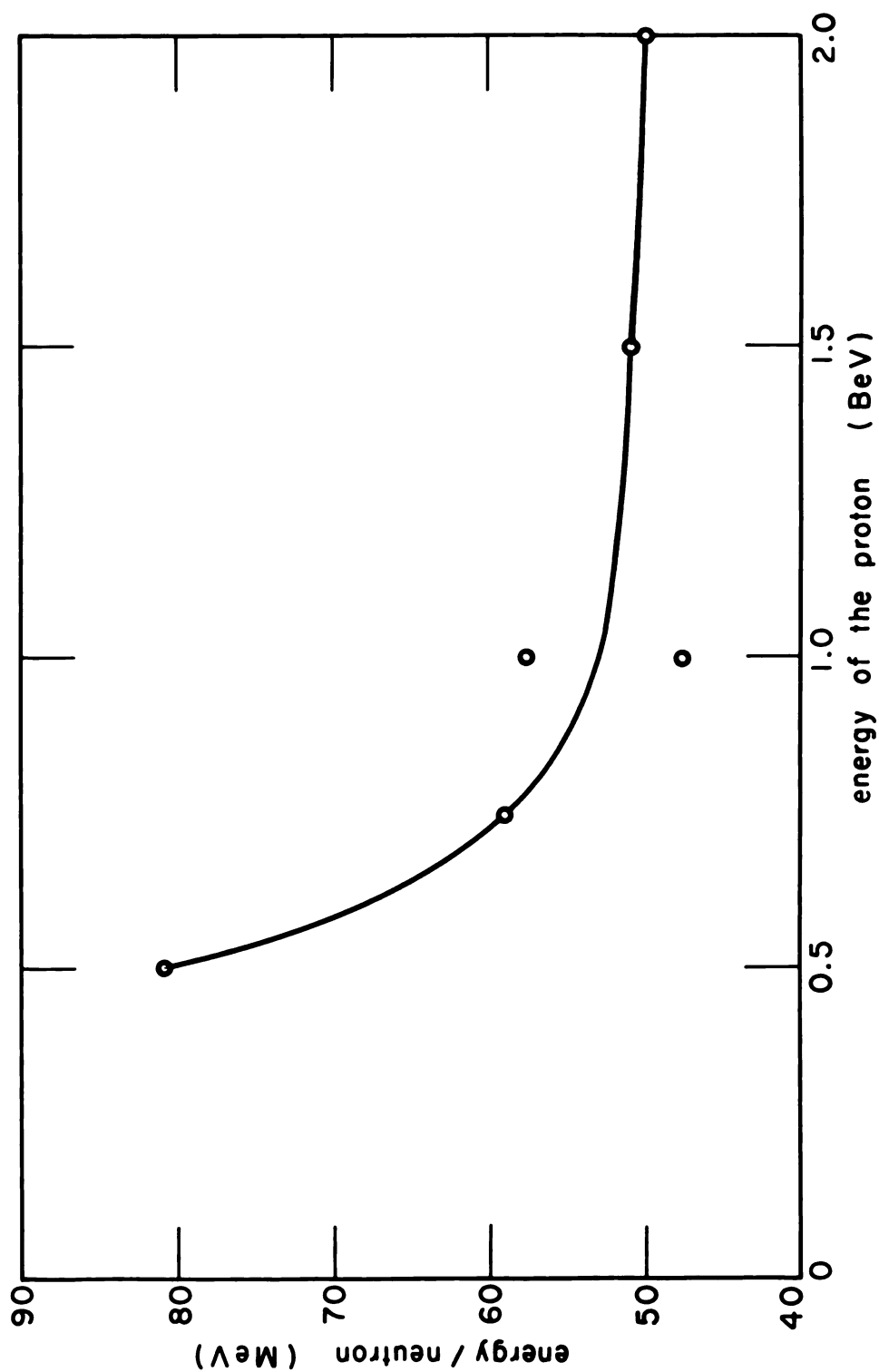
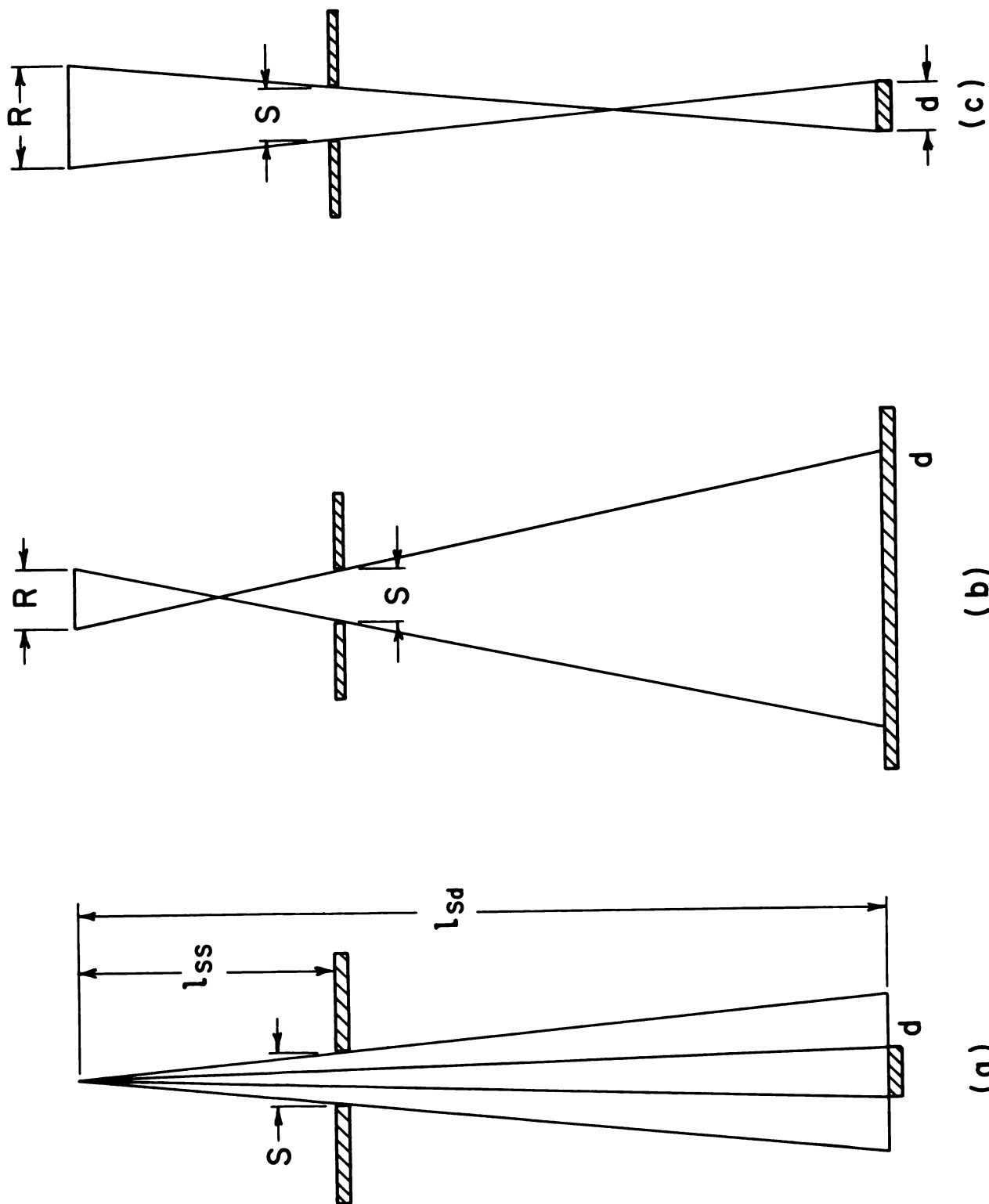


Figure 2. The energy required to liberate a low energy neutron by bombarding a thick lead target with high energy protons as a function of the energy of the incident proton.



(a)

(b)

(c)

Figure 3. Idealized geometrical arrangements for three different configurations used for neutron spectrometers.

detector, l_{ss} for the distance between the source and the slit, and l for the distance between the slit and the detector. If we ignore all other considerations, such as background, and place the defining slit halfway between the source and the detector, then the source and the detector should be the same size. If, however, the defining slit is much closer to the source, as it usually is to reduce background, we have two cases we must examine. In the first case the detector is so large that the defining slit effectively determines the solid angle of the neutron beam. In this case the intensity at the detector is given by the following formula:

$$I = \frac{B h s}{2 l_{ss} l_{sd}} \cdot R$$

where B is the source brightness and h is the height of the source. We see then that the intensity at the detector is directly proportional to the total number of neutrons emitted by the source.

However, if we look at the more usual case, where the defining slit is much closer to the source, then the intensity of the detector is given by the following formula:

$$I = \frac{Bh}{2 l_{ss} l_{sd}} \cdot \left(s + \frac{l_{ss}}{l_{sd}} d \right)$$

We can see that, in the first approximation, the effective width of the source is the width of the defining slit.

In the Columbia neutron velocity spectrometer, the detector is 120 cm wide and 30 cm high. The source-defining slit is 6 meters from the source, and the source detector path length is 200 meters; the width of the source defining slit is 20 cm. Consequently, the useful length of the source is about 24 cm, which is approximately what is now used in the Columbia neutron velocity spectrometer. In Figure 4 the range of protons in lead in centimeters is plotted as a function of the energy of the incident proton. From this graph we can see that the range increases quite rapidly as a function of energy. In the energy range from a few MeV to 200 MeV, the range increases as $E^{1.8}$. Above 200 MeV the exponent in energy dependence formula decreases and, as one can see from this graph, above 10 BeV the exponent is less than one. However, it is not the energy dependence of the range in which we are interested, but the size of the neutron emitting target.

The neutron flux profile of a 1-BeV proton beam striking a 24-inch lead target with the neutrons moderated by D_2O is shown in Figure 5, also taken from the same ORNL progress report. We can see from these results that the length

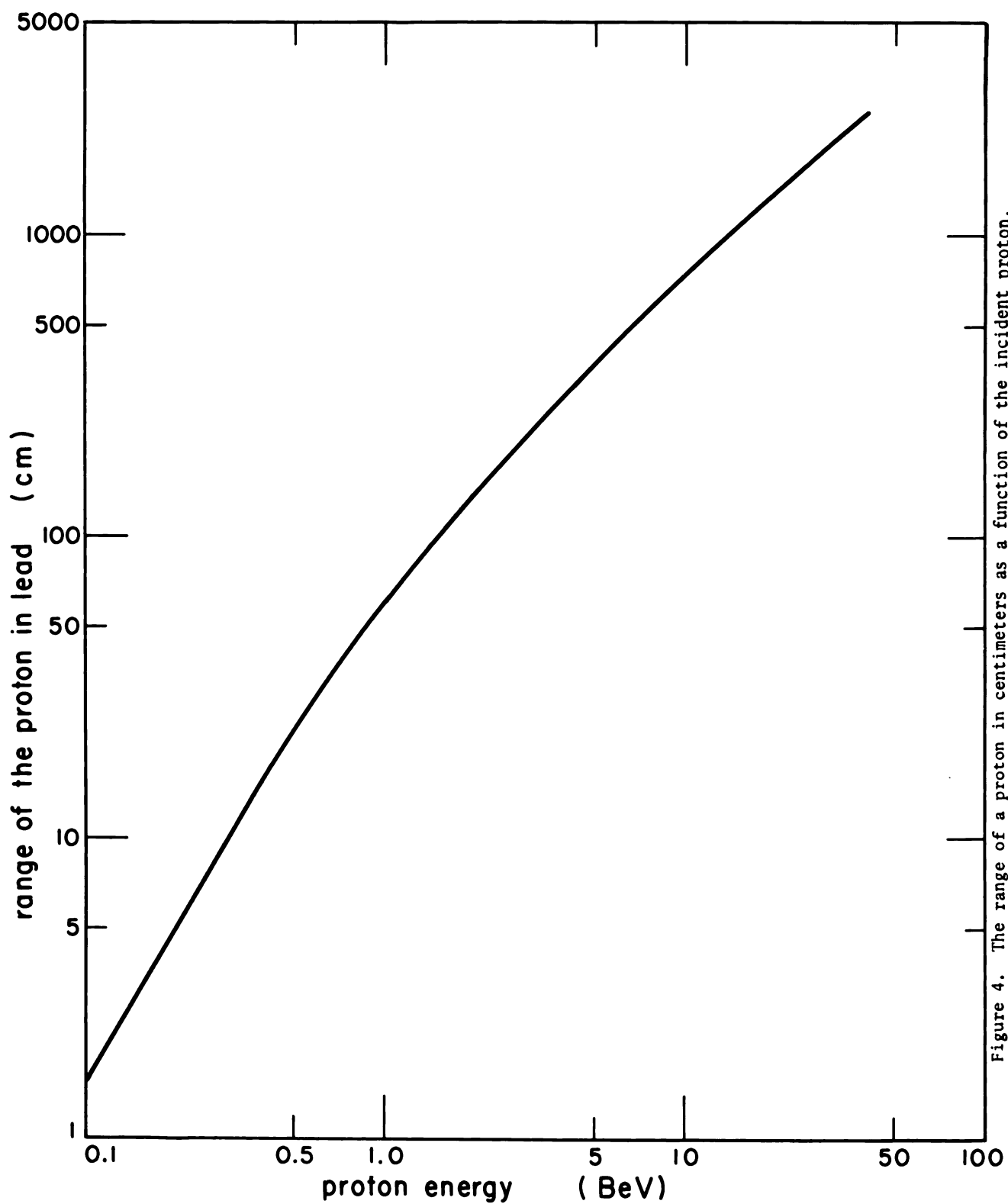


Figure 4. The range of a proton in centimeters as a function of the incident proton.

of the moderated neutron source is approximately equal to the length of the target. Therefore, the source-defining slit in the system will effectively determine the length of the target or the range of the proton which can be used for neutron time-of-flight spectroscopy. I have presented this material assuming that the neutron production was uniform, spherically symmetrical and that the beam was emitted perpendicular to the direction of the incident proton beam. One might argue that if the neutron beam is emitted parallel or anti-parallel to the incident proton beam, then the intensity would increase in direct proportion to the total number of neutrons produced. However, the average energy of the useful neutrons is about 4 MeV, since they are evaporation neutrons and, consequently, the neutron beam will be attenuated in the targets. Hence, viewing the neutron beam parallel or anti-parallel to the direction of the proton beam does not improve the effective source intensity. If the neutron beam is emitted parallel to the direction of the proton beam, then the very high energy neutron background from knock-ons is troublesome and, although we have assumed spherical symmetry, the symmetry is not spherical nor is the neutron production uniform and the number of neutrons emitted in the backward direction is less than the number emitted in the forward direction. However, taking account of these additional details does not substantially alter the situation.

I do not think one would want to use a much larger detector than the one used at the Columbia neutron velocity spectrometer. In addition, a 1-BeV proton travelling at approximately the velocity of light requires about 2 nanoseconds to traverse a lead target and the duration of the pulse will increase as the energy of the proton increases. Consequently, I think we can conclude that it is unlikely that one would want to use a proton beam with energies higher than 1 BeV for neutron time-of-flight spectroscopy. There are other considerations about synchrotrons to be given later which also support this conclusion.

3. CRITERIA FOR COMPARISONS

In order to compare cyclic particle accelerators as pulsed neutron sources with other types of pulsed neutron sources, one has to establish some criteria for comparisons. In the past, many scientists have attempted to establish a single criterion as the "figure of merit" of a neutron velocity spectrometer. The figure of merit has been defined in various ways over the years in an attempt to compare choppers at reactors and accelerators and, depending upon your prejudice, the figure of merit has been unfair to either one or the other because dissimilar experiments are and should be performed with different devices due to the different characteristics of the different

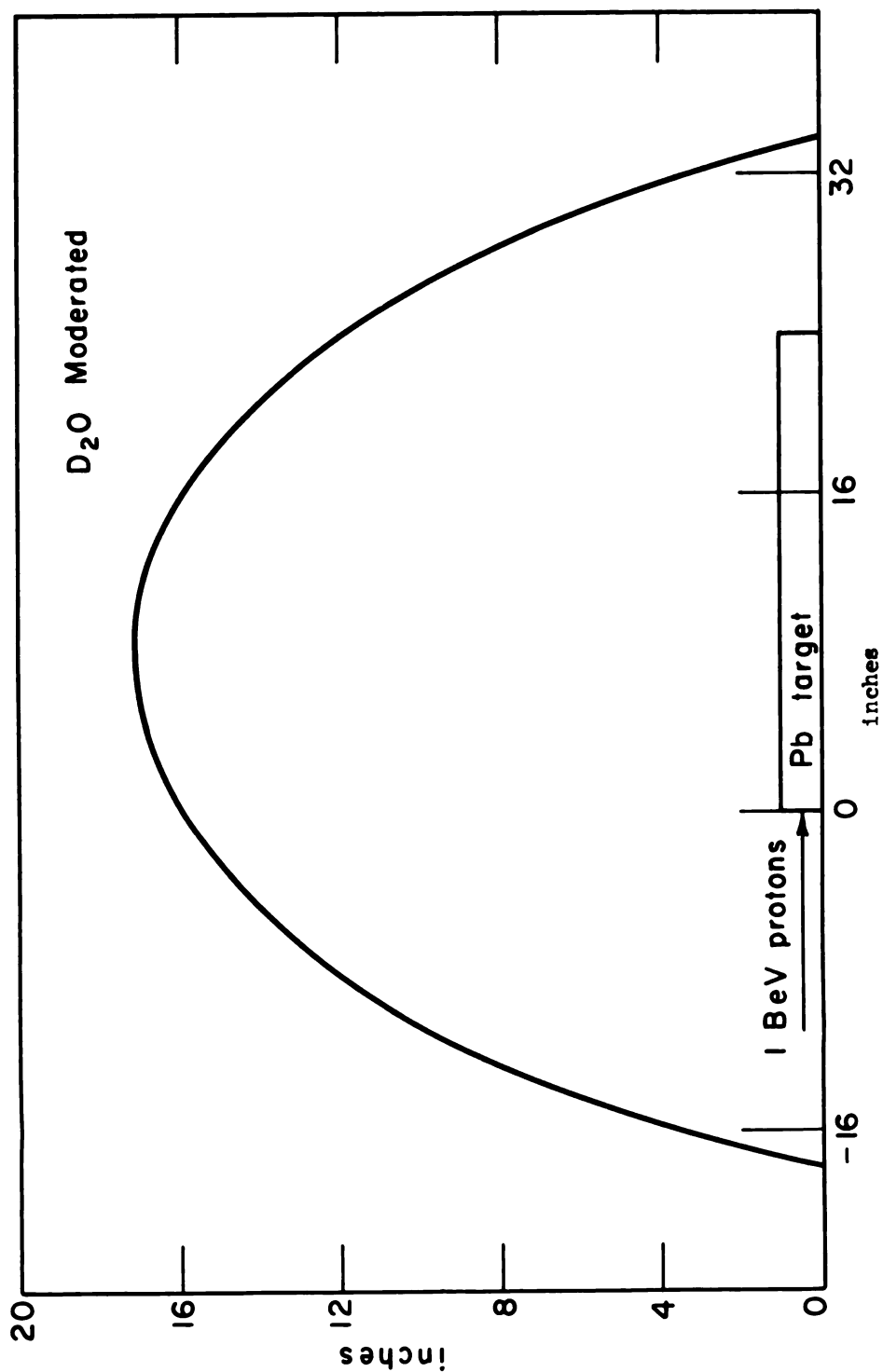


Figure 5. The thermal neutron flux profile produced by a 1 BeV 65 mA proton beam striking a 24 inch lead target with the neutrons moderated by D₂O. The area inside the curve corresponds to the volume for which the flux is $\geq 10^{16}$ n/cm²/sec.

devices. The figure of merit has even included, at times, the fraction of time a particular accelerator was used for neutron time-of-flight work in order to show the particular installation in a relatively unfavorable light.

Unfortunately, the criteria which should be used are not clear because of the different characteristics of the various devices. These characteristics cannot be compared directly, since this would be like comparing apples and oranges. It is fair to use a single figure of merit only when comparing two devices which do the same type of experiment. In my opinion, one figure of merit is not a good criterion to use when comparing devices which do different types of experiments.

In my opinion, two main criteria must be used to compare the merits of pulsed neutron sources. The first criterion is the maximum intensity in the burst. The second criterion is the total number of neutrons produced per second in the energy interval of interest when the device is used as a neutron velocity spectrometer. It is the burst intensity (assuming suitable burst width) which determines what experiments can be done because this factor determines the signal-to-noise ratio; the total number of neutrons produced per second determines the length of time required to do an experiment. To illustrate this, consider the nuclear explosion source, which has by far the highest burst intensity (thus permitting unprecedented measurements of capture cross sections of short-lived radioactive isotopes, etc.). On the other hand the nuclear explosion has an extremely low "repetition rate"--once every few months. Thus, the average number of neutrons per second may be rather small and therefore certain experiments may still take a long time to complete. Based on these considerations, I suggest the above two criteria for determining the merit of a pulsed neutron source when used as a neutron velocity spectrometer.

4. PULSED NATURE OF CYCLIC PARTICLE ACCELERATORS

All cyclic particle accelerators which use radio frequencies to accelerate particles across many gaps are inherently pulsed devices. The pulsed nature of the accelerator arises from the fact that the particle must receive a certain amount of energy per radio frequency cycle in order to remain in the proper phase to be accelerated. If the particle gains too much or too little energy, then it will be decelerated by the radio frequency and be returned to the proper phase. This, in essence, is the principle of phase stability, which was discovered independently by McMillan and Veksler in 1944.^{6,7} I include all cyclic particle accelerators because I want to include in my evaluations the Los Alamos Meson Physics Facility, in addition to attempting to evaluate cyclotrons, synchrocyclotrons, and synchrotrons as pulsed neutron sources.

A great deal of work has been done to determine the phase acceptance angle from the ion source in the radio frequency cycle for many cyclotrons, synchrocyclotrons, and synchrotrons. The acceptance angle varies with the central magnetic field, the electrostatic and magnetic focusing configuration, the rate of change of frequency, and the detector injector conditions. I do not want to spend much time describing the exact conditions which govern the acceptance angle, but I think it is simple to see that, roughly, the acceptance angle is dependent on the accelerating voltage of the radio frequency electrode. The higher the accelerating voltage, the higher the energy gain per accelerating gap with a consequent more intense microstructure for the beam of the accelerator. In ordinary cyclotrons and isochronous cyclotrons, the accelerating voltage should be as high as possible in order to obtain the maximum current in the beam. The desire of the experimenter to eliminate the microstructure caused by rf pulses influences the accelerator designer to reduce the rf voltage with a consequent decrease in beam intensity. However, if an accelerator were to be designed for pulsed use for time-of-flight measurements, then the rf accelerating voltage should be as high as possible.

In synchrocyclotrons and synchrotrons, the accelerating voltages are considerably lower than for cyclotrons and isochronous cyclotrons. Therefore, in synchrocyclotrons and synchrotrons, the proton pulse is more spread out, both in angle and in time, than in cyclotrons. In synchrocyclotrons, the rate of change of frequency determines the number of particles accepted from the ion source and the maximum voltage which can be applied to the dees of the accelerator. Roughly, the wider the frequency range which must be covered, the longer will be the pulse length of the microstructure. In synchrotrons, the injection is made as long as possible in order to capture the maximum number of particles in a stable orbit. For the Alternate Gradient focusing Synchrotron, the injection is for one complete turn when the frequency is 1.5 Mc. Therefore, the beam has a macrostructure of the order of 10 microseconds, determined by the total time the injector feeds ions into the accelerator and a microstructure determined by the radio frequency which, according to Dr. Blewett, is 15 to 20 nanoseconds wide.

If the details of the microstructure and macrostructure of the beam of existing synchrotrons are examined, one can see that the maximum use of the principle of phase stability which enables synchrocyclotrons and synchrotrons to operate lengthens the microstructure pulse and makes them less useful as pulsed intense neutron sources. Therefore, we can conclude from the considerations of beam dynamics and the range of the protons in the target that synchrotrons as presently designed will not be useful as pulsed intense neutron sources.

For almost all cyclotrons and isochronous cyclotrons, the acceptance angle of the beam from the arc source appears to be of the order of 5 per cent. Cierjacks and Beckurts, in a paper presented at the International Conference on the Study of Nuclear Structure with Neutrons last year at Antwerp,⁸ estimated the pulse length of the microstructure pulse in a large number of cyclotrons in Canada, France, Germany, Italy, the Netherlands, the USSR, the UK, and the USA. The minimum duration of the ion pulse listed in Cierjacks and Beckurts' Table is 1.5 nanoseconds for the Karlsruhe isochronous cyclotron which has been designed for high Dee voltages and the maximum pulse duration is 4.8 nanosecond for the Orsay isochronous cyclotron which has been designed for fairly low accelerating voltage. Langsford, et al.,⁹ from Harwell, in a paper submitted to the same Conference, have estimated that the duration of the pulse in the Harwell synchrocyclotron is about 9 nanoseconds. The duration of the pulse in the Columbia synchrocyclotron has been measured to be 20 nanoseconds. The microstructure of the Los Alamos Meson Physics Facility is expected to have a pulse duration of 0.25 nanoseconds every 4.5 nanoseconds, with a macrostructure of pulse duration 500 microseconds every 8,300 microseconds. From this information, the maximum current in the pulse can be determined and, knowing the energy of the accelerator, the maximum neutron production rate in the pulse can also be obtained, which is one of the criteria I introduced as determining what experiments can be done.

5. EXPERIMENTAL PROGRAM

The experiments which will be performed using these accelerators as pulsed high intensity neutron sources have in the past been divided into two categories: 1) slow neutron time-of-flight spectroscopy and, 2) fast neutron time-of-flight spectroscopy. Several years ago entirely different techniques were used in these two fields. Slow neutron spectroscopists used moderated neutrons, a neutron pulse of several microseconds duration and source detector path lengths of many meters, whereas fast neutron spectroscopists used unmoderated neutrons, a neutron pulse of nanoseconds duration and path lengths of the order of one meter. However, as source intensities have increased and other techniques have improved, the duration of the pulses used by slow neutron spectroscopists have decreased to the nanosecond range and the source detector distances used by fast neutron spectroscopists have increased to several tens of meters. The energy range studied by both groups now overlap. In fact, it has been just such an overlap that has been desired for many years and we are delighted that the overlap now exists. However, because of this overlap we now need different criteria to distinguish between the two types of neutron time-of-flight spectroscopy.

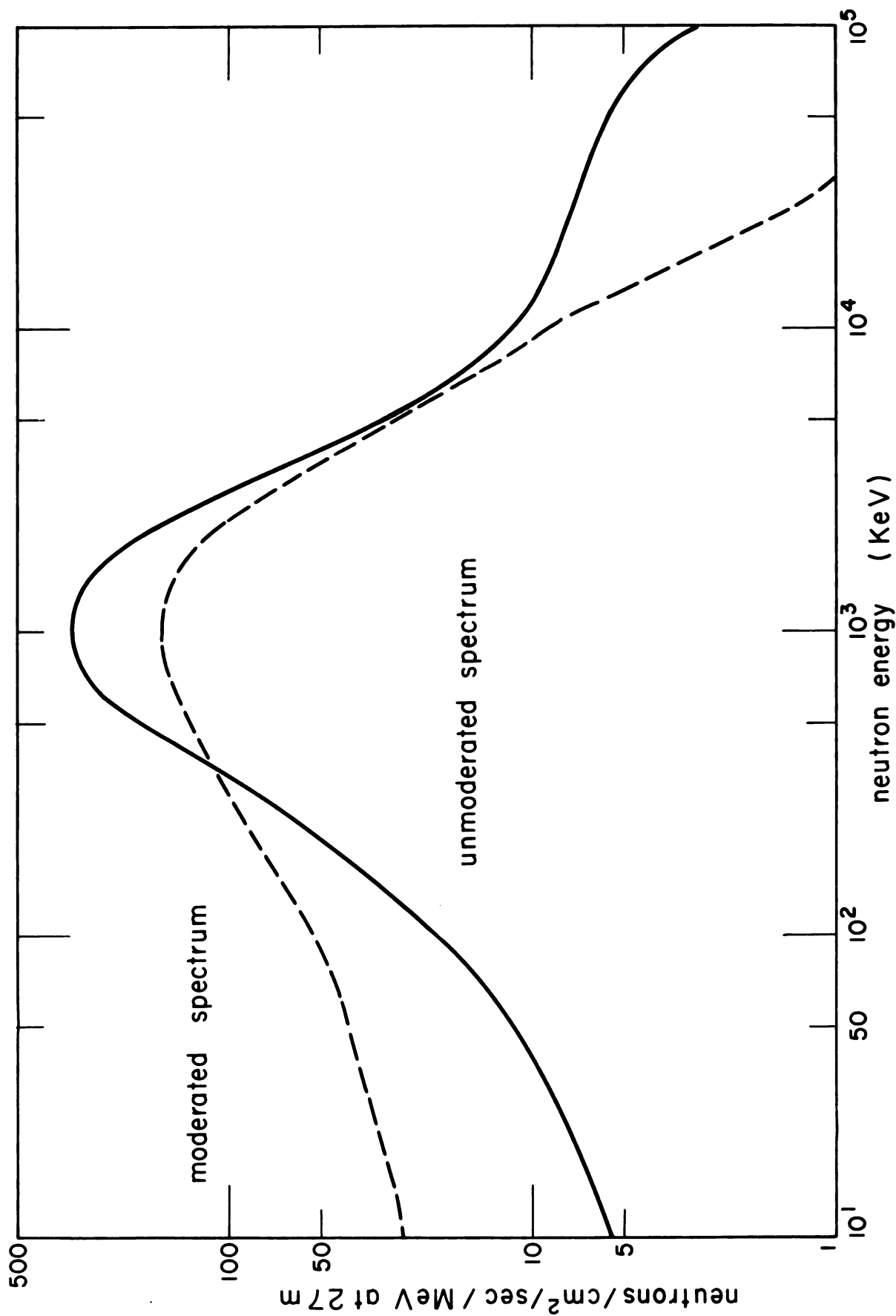


Figure 6. The energy spectra of neutrons emitted by a target bombarded with 150 MeV protons with the ordinate in neutrons/cm²/sec per MeV at 27 meters. The solid line indicates the unmoderated neutrons and the broken line indicates an arrangement with 2.5 cm of polyethylene between the source and the detector.

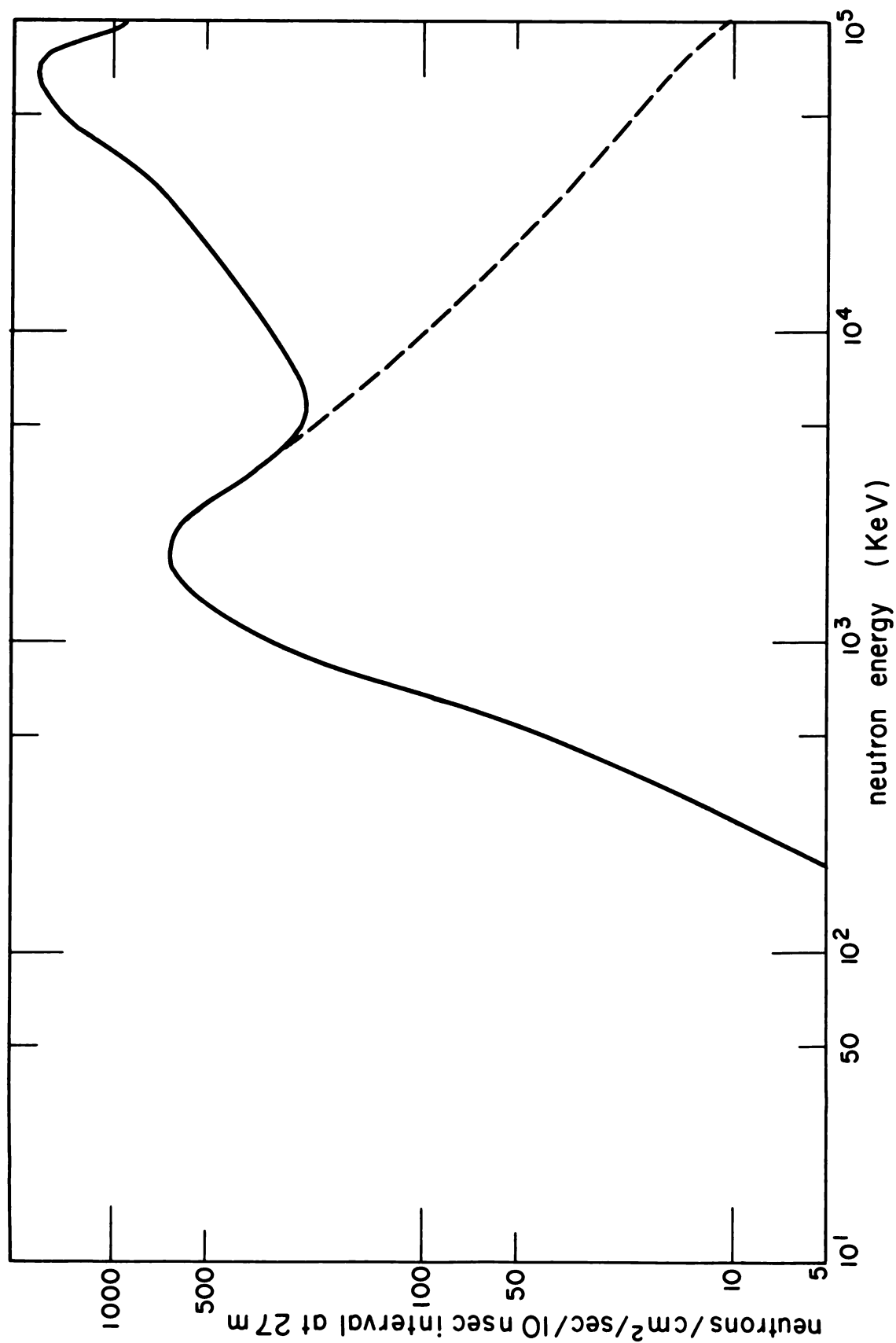


Figure 7. The energy spectra of neutrons emitted by a target bombarded with 150 MeV protons with the ordinate in neutrons/cm²/sec/10 nanosecond intervals at 27 meters. The solid line indicates the unmoderated neutrons and the broken line indicates an arrangement with 2.5 cm of polyethylene between the source and the detector.

In my opinion, the principal difference between slow and fast neutron time-of-flight spectroscopy is in the energy response of the detector used. In slow neutron time-of-flight spectroscopy, the detector responds to neutrons which have energies as low as can be obtained in the laboratory and the efficiency of the detector usually increases as $\frac{1}{v}$. In fast neutron spectroscopy, the detector will not respond to neutrons with energies below some minimum energy and, in particular, will not respond to thermal neutrons. Therefore, systems which are used for fast neutron spectroscopy can have a repetition rate as high as a few million per second, whereas systems used for slow neutron spectroscopy cannot have a repetition rate of more than a few hundred per second. The allowable repetition rate used in fast neutron spectroscopy depends on the source detector path length and the minimum energy of the neutron to be measured. All slow neutron velocity spectrometers can be used as fast neutron spectrometers but fast neutron spectrometers designed to operate with high repetition rates have higher counting rates in the few hundred keV region because the average counting rate is directly proportional to the repetition rate. Therefore, at some energy which now is a few hundred keV, the apparatus used for slow neutron spectroscopy cannot compete with the apparatus used for fast neutron spectroscopy.

It is interesting to note that another of the original differences between slow and fast neutron spectroscopy, the use of moderated neutrons, has also almost disappeared. Langsford et al.⁹ have measured the spectrum in the forward direction of unmoderated neutrons and neutrons moderated by a 2.5 cm thick polyethylene slab. The moderated and unmoderated neutron spectra are shown in Figure 6 where the ordinate is the flux in neutrons/cm²/sec/MeV at 27 meters. In Figure 7, the same spectra are shown with the ordinate in neutrons/cm²/10 nanosecond intervals at 27 meters, which is the ordinate of interest for velocity spectrometer work. The solid line is the unmoderated spectrum and the broken line is the moderated spectrum. Inspection of these graphs shows that above 200 keV, the flux from the unmoderated target is higher than from the moderated target and, therefore, for measurements above 200 keV, an unmoderated target should be used.

Recently Bartholomew, Milton, and Vogt² defined a figure of merit, M, for neutron spectrometers as follows:

$$M = \frac{Q\tau f}{\tau^2}$$

where Q is the maximum intensity in the pulse, τ is the pulse duration in seconds, and f is the repetition rate. $Q\tau f$ is the number of neutrons produced per second. Since this figure of merit is directly proportional to the

TABLE I

Accelerator	Energy in Mev	Average Current in Microamps	Average Neutron Intensity in Neutrons per Sec.	Repetition rate in Pulses per Sec.	Burst Width in Nanosecs.	Peak Current in Milliamps	Bartholomew Figure of Merit	Instant. Neutron Intensity*	Average Neutron Intensity* as a Spectrometer
Columbia Synchro-cyclotron	400	0.5	2.4×10^{13}	60	25	330	4×10^{28}	1.6×10^{19}	2.4×10^{13}
Harwell Synchro-cyclotron	150	1.0	10^{13}	200	10	500	10^{29}	5×10^{18}	10^{13}
Karlsruhe	50(d)	100	6×10^{14}	20,000	1	150	1.2×10^{31}	6×10^{17}	2×10^{13}
Oak Ridge	100	100	3×10^{14}	continuous	2.2				
Improved Columbia Synchro-cyclotron	600	50	5×10^{15}	300	10	13,200	1.9×10^{31}	6.4×10^{20}	5×10^{15}
Improved Karlsruhe	50(d)	100	6×10^{14}	200,000	1	150	1.2×10^{32}	6×10^{17}	2×10^{14}
LAMPF (Slow Neutron)	1,000	1,000	1.2×10^{17}	120	0.25	~1,000	6×10^{31}	1.2×10^{20}	~ 10^{12}
LAMPF Fast Neutron	1,000	1,000	1.2×10^{17}	12,000	0.25	~1,000	6×10^{33}	1.2×10^{20}	~ 3.6×10^{15}
SOC Slow Neutron	1,000	75,000	9×10^{18}	400	2.5	7.5×10^6	4×10^{37}	2.4×10^{26}	2×10^{18}
SOC Fast Neutron	1,000	75,000	9×10^{18}	200,000	2.5	3.75×10^3	4×10^{37}	1.2×10^{23}	2×10^{18}

* neutrons/second

repetition rate, it favors a system designed for fast neutron spectroscopy. It is a good figure of merit to use to compare fast neutron spectrometers and to compare slow neutron spectrometers, but should not be used to compare slow with fast neutron spectrometers.

The characteristics of several accelerators for neutron velocity spectrometer experiments which are or could be used as neutron velocity spectrometers are given in Table 1. Also included are optimistic estimates of improvements which can be made in existing accelerators and some future accelerators. For the purposes of neutron spectroscopy, the most important numbers are the last columns which give the instantaneous burst intensity and the average neutron intensity for the accelerator when used as a neutron velocity spectrometer. The instantaneous burst intensity is obtained by multiplying the number of protons per second incident on the target for the peak current by the number of neutrons produced per proton. The average neutron intensity is obtained by multiplying the peak intensity by the burst width and the number of pulses per second which can be used for neutron spectroscopy.

The last two columns of the table show that among existing accelerators, the Columbia synchrocyclotron has the highest intensity in the burst and the highest average intensity when used as a neutron spectrometer. The accelerator with the highest average neutron intensity is the Karlsruhe isochronous cyclotron. One might expect that I would produce a table which results in showing the Columbia synchrocyclotron neutron velocity spectrometer as the best for the criteria I have chosen. However, I want to point out that, by Bartholomew's criterion, the Karlsruhe cyclotron has a much higher figure of merit because this criterion favors fast neutron spectrometers. To properly compare accelerators, we must separate the fast from the slow neutron spectrometers.

From the repetition rates given in Table 1, the Columbia synchrocyclotron, the Harwell synchrocyclotron, and the Los Alamos Meson Physics Facility can be used as slow neutron time-of-flight spectrometers. I have also indicated that the separated orbit cyclotron can be used as a slow neutron time-of-flight spectrometer with both very high burst and average intensity. Of course, no one knows exactly how one would extract the pulsed beam from the separated orbit cyclotron in the proper way and I have assumed, in order to obtain the high burst and average intensity, that the beam from the separated orbit cyclotron could be stored in a storage ring and deflected into the target in the time required for one microstructure pulse. The decrease in the intensity for the separated orbit cyclotron when used as a neutron velocity spectrometer from the average intensity comes from the fact that the separated orbit

cyclotron will probably use a much higher rf than the synchronous frequency of the proton in the magnetic field. Therefore, there will be several microstructure pulses and all but one of these will have to be eliminated in the same manner as the extraneous microstructure pulses were removed in the Karlsruhe pulsed isochronous cyclotron. I do not think it is any more speculative to assume that one can pulse the separated orbit cyclotron appropriately than it is to assume that one can build a separated orbit cyclotron.

The existing and improved Karlsruhe cyclotron, the Los Alamos Meson Physics Facility, and the separated orbit cyclotron can be used as fast neutron spectrometers. Here again I have assumed that the pulsed beam can be extracted for the LAMPF and SOC accelerators in an appropriate way. In the SOC, only the extraneous microstructure pulses have been eliminated and in the LAMPF, one 0.25 μ sec microstructure every 500 nanoseconds of the macrostructure pulse was assumed useful for fast neutron spectroscopy.

From the results of Table 1 we can see that the improved Columbia synchrocyclotron will be by far the best slow neutron spectrometer. (The improvements in the Columbia synchrocyclotron are described in a report by J. Rainwater¹⁰ which sets forth the proposed modification program of the synchrocyclotron and it is admittedly optimistic.) The LAMPF would not be a very useful slow neutron spectrometer, but would be an excellent source for fast neutron spectroscopy. Of course, the SOC appears to be the best from all points of view, but its operation is rather nebulous.

Since all the accelerators I have discussed are expected to be used for experiments, I would like to end this talk by showing some data. Figure 8 shows the results on vanadium taken with the Columbia synchrocyclotron neutron velocity spectrometer and Figure 9 shows the results for iron. The resolution is about 0.5 nanoseconds/meter as measured by the width of the resonances. The very impressive results on iron of Cierjacks, Forte, Kropp and Ursel¹¹ using the Karlsruhe cyclotron are shown in Figure 10. The resonances at 511.8 and 513.3 keV have been separated and show the resolution to be about 0.04 nanoseconds per meter.

The principal conclusion of this paper is that cyclotrons and synchrocyclotrons have proven to be very useful devices as pulsed intense neutron sources. If the pulsed characteristics of a cyclic particle accelerator are maximized in the design of an accelerator, enormous improvements can be expected.

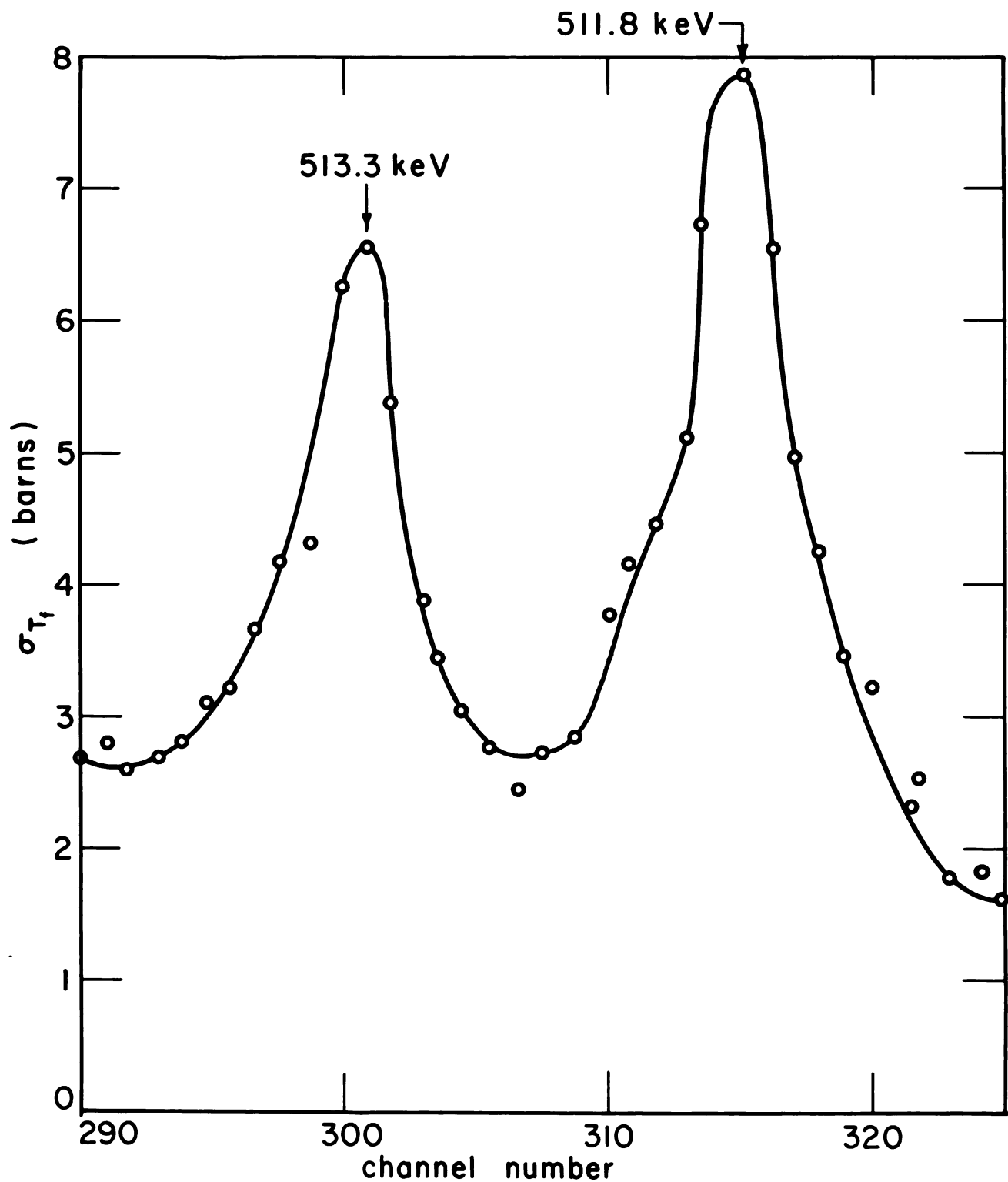


Figure 10. The total neutron cross section of iron measured with the Karlsruhe neutron spectrometer(Ref. 8)

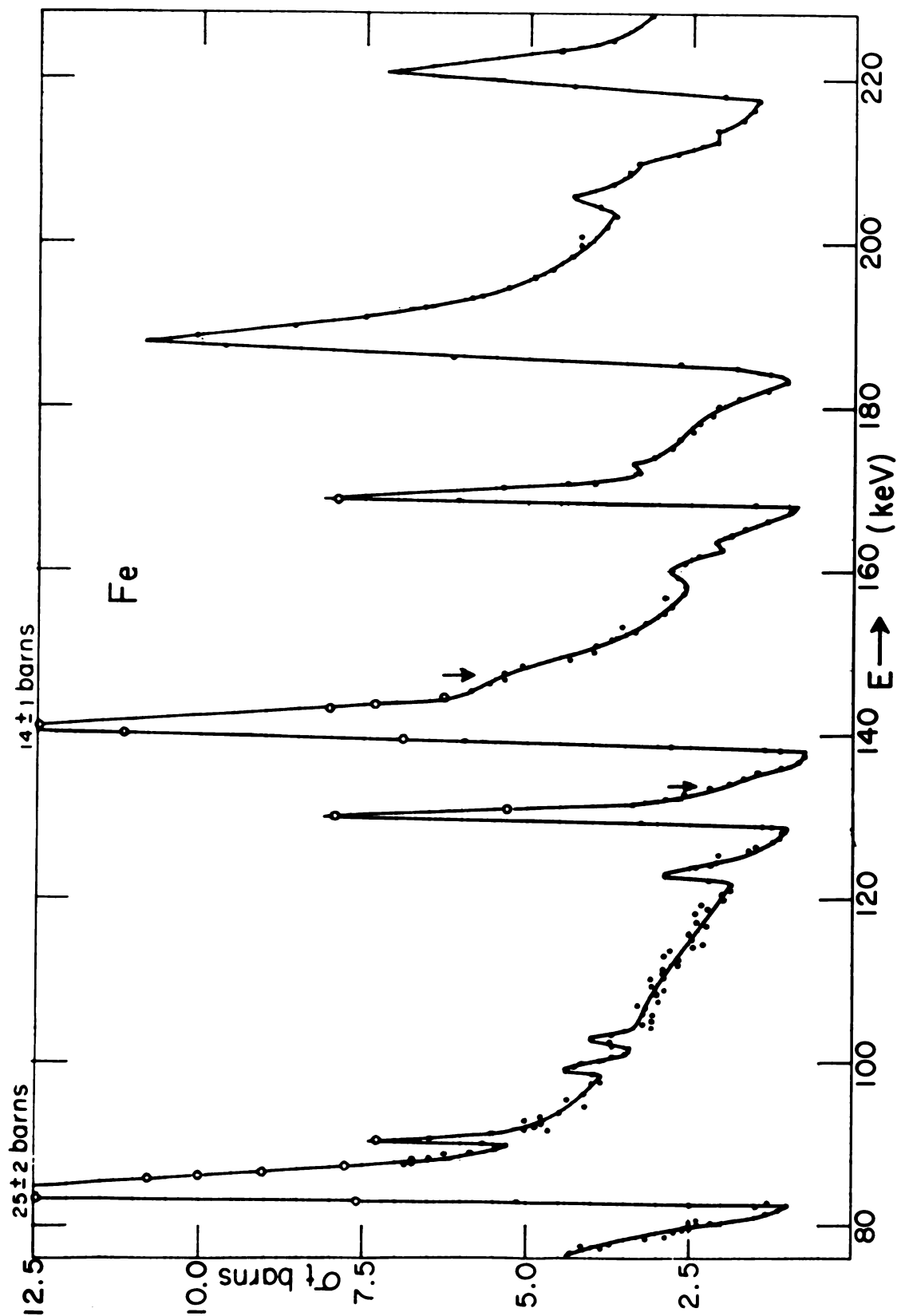


Figure 9. The total neutron cross section of iron measured using the Columbia Synchrocyclotron Neutron Velocity Spectrometer.

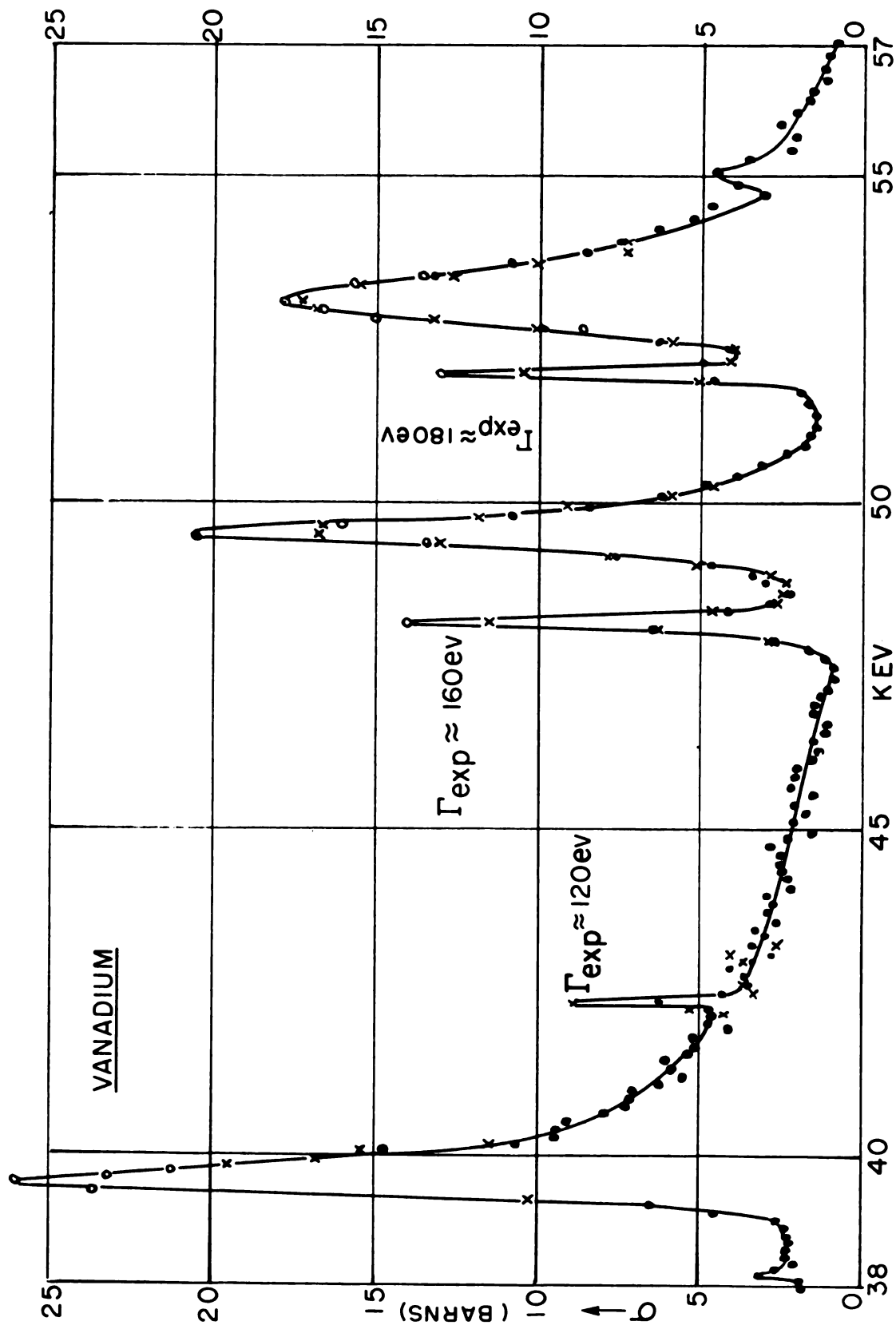


Figure 8. The total neutron cross section of vanadium measured using the Columbia Synchrocyclotron Neutron Velocity Spectrometer.

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THE SECTOR-FOCUSED CYCLOTRON AS A POWERFUL TOOL FOR FAST NEUTRON TIME-OF-FLIGHT RESEARCH

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A b s t r a c t: At the Karlsruhe Isochronous Cyclotron a fast neutron time-of-flight spectrometer has been put into operation and the first measurements have been carried out. The performance data of this apparatus are as follows: average unmoderated 4π neutron output per second $I_a = 2 \cdot 10^{13}$ neutron/s at maximum, neutron pulse width $\tau = 1 \pm 0,3$ ns (f.w.h.m.); pulse repetition rate 20 kc/s and length of the flight path 57 m at present. The figure of merit $M = I_a/\tau^2$ thus is about $1 \cdot 10^{31} \text{ s}^{-3}$. Possible improvements for the spectrometer are discussed. In addition general aspects of the use of sector-focused cyclotrons as very intense pulsed neutron sources are considered; in particular, neutron production is discussed in relation to cyclotron design features. Finally, an outline of possible experiments is given for which very intense fast neutron time-of-flight devices are advantageous.

1. Introduction

In a recent paper ¹⁾ some general aspects of the use of sector-focused cyclotrons as very intense pulsed neutron sources for fast neutron time-of-flight spectroscopy were pointed out. In the mean time the neutron time-of-flight facility at the Karlsruhe isochronous cyclotron has been put into operation and first transmission measurements on some light elements have been made. The importance of a sector-focused cyclotron time-of-flight facility for such experiments could be demonstrated by some first physical results.

Neutron time-of-flight experiments are greatly facilitated by the high average beam current and the small pulse width (a few nanoseconds) of the microstructure pulses from sector-focused cyclotrons. With these machines beam intensities at relativistic energies are enhanced by a

factor of about one hundred compared with synchrocyclotrons, which operate in the same energy range (10 to several hundred Mev). In addition, the microstructure pulse widths in sector focused cyclotrons are smaller at maximum by about a factor of ten. However, isochronous machines run continuously with a microstructure pulse repetition rate of 10 - 30 Mc/s which is far too high in view of frame overlap problems in high resolution time-of-flight experiments. Reducing the repetition rate by deflection of single microstructure pulses would be an undesirable sacrifice in intensity. At the Karlsruhe isochronous cyclotron a novel "deflection-bunching" system is now in use which avoids the frame overlap problem while largely preserving the high average neutron intensity available from the internal beam. This system reduces the repetition rate from 33 Mc/s to 20 kc/s while the average neutron intensity is reduced by only a factor of 30 at present. But this is not a true limit. In sector-focused cyclotrons one should, in principle, be able to "bunch" the entire average beam intensity into bursts of a few nanoseconds duration. This gain in neutron intensity per second, however, will be obtained at the expense of higher pulse recurrence frequencies. The deflection-bunching system and the neutron time-of-flight spectrometer at the Karlsruhe isochronous cyclotrons will be described in detail in section 2 of this paper while in section 3 more general aspects of the use of sector-focused cyclotrons will be discussed. In table I some characteristic data for sector-focused cyclotrons which are in use now are shown ²⁾.

Table I Isochronous Cyclotrons and their Characteristics

		Energy MeV	Int. Beam Current μA	Dees	R.F. Mc/s	Ion pulse .. length ns
Canada	Winnipeg	4.90 H ⁺	100	2 45°	14 - 29	1.7
France	Orsay	4.70 α ⁺	500	1 180°	4 - 10.5	4.8
	Saclay	4.29 p ⁺	100	1 180°	5 - 23.5	2.2
Germany	Karlsruhe	50 α ⁺	100	3 60°	33	1.5
Italy	Milan	4.45 p ⁺	100	1 180°	17 - 22	2.3
Netherlands	Delft TH	12 p ⁺	500	1 180°	21	2.4
	Fres Univ.	4.35 p ⁺	100	1 180°	5 - 22	2.3
	Philips-Duphar	4.28 p ⁺	500	1 180°	10 - 21	2.4
	Philips Lab.	4.30 p ⁺	1000	1 180°	5 - 23	2.2
U.K.	Amersham	4.25 p ⁺	1000	1 180°	10 - 20.6	2.4
	Birmingham	4.12 p ⁺	1000	1 180°	7 - 16	3.1
	Harwell	4.90 p ⁺	500	1 180°	7.6 - 22	2.3
	Rutherford	3 p ⁺	1000	1 180°	15 - 20	2.5
USA	Michigan SU	4.55 p ⁺	1000	2 134°	13.8 - 21.8	2.3
	Oak-Ridge HL	4.70 p ⁺	500	1 180°	7 - 22.5	2.2
	Pomona C	2 p ⁺	25	1 180°	13.9	3.4
	UC Berkeley	18 He ³	20	2 55°	17	3.0
	UC Berkeley	4.60 p ⁺	1000	1 180°	5.5 - 16.5	3.0
	UC Los Angeles	4.50 H ⁺	5	2 45°	26.5 - 29	1.7
	U Colorado	4.50 p ⁺	1000	1 180°	7.5 - 21.3	2.4
	U Michigan	4.37 p ⁺	500	2 180°	6 - 16	3.1
	U Illinois	4.15 p ⁺	500	1 180°	9 - 18.5	2.8
	U Washington	4.29 p ⁺	500	1 180°	7 - 21	2.4
USSR	Alma-Ata	4.40 p ⁺	400	2 180°	---	-
	Dubna	14 α ⁺	20	1 180°	10.5	4.8
	Leningrad	4.8 p ⁺	20	1 180°	9 - 20	2.5

* Also other ions

** Calculated for the highest radio-frequency applied, assuming a duty cycle 5%

2. The Karlsruhe Sector-Focused Cyclotron as a Pulsed Fast Neutron Source for Time-of-Flight Experiments.

Deflection-Bunching System: The principle ³⁾⁴⁾ of operation is shown in fig. 1. In the normal continuous operation of our isochronous cyclotron there are three ion bunches delivered from the source per cycle because acceleration is accomplished in the third harmonic mode. The deflection bunching system consists of two coupled electrostatic deflectors, both of which are used for axial beam deflection. One deflector which is located near the center of the machine is used for a twofold purpose:

- to eliminate two out of three microstructure pulses by deflection to a beamstop and
- to form ion bunches of 4,5 μ sec duration (each consisting of fifty microstructure pulses) with a repetition rate of 20 kc/s.

Both conditions can be accomplished by an appropriate sine wave which is indicated schematically in the upper part of fig. 1.

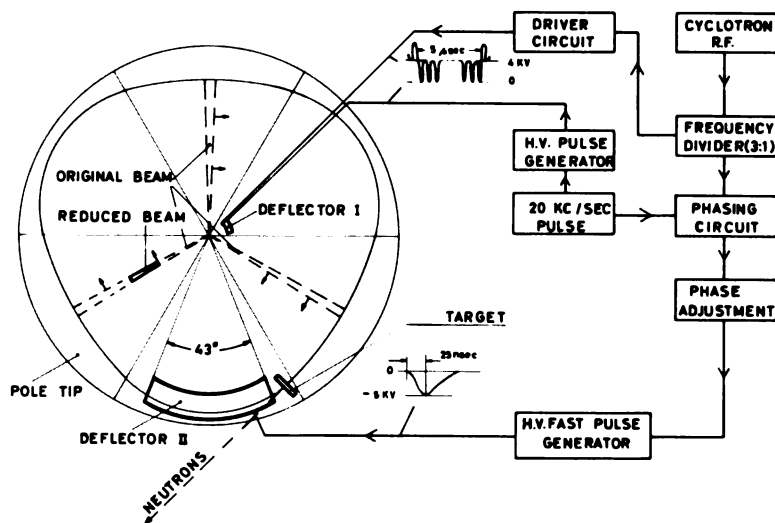


Fig.1 Scheme of the bunching deflection system.

A second pair of deflector plates is located at a mean radius of 39 inches and serves to simultaneously deflect the whole set of microstructure pulses to a target positioned above the median plane of the cyclotron. At the time of the deflection the bunch is distributed over 4 inches in radius. Before striking the target the deflected beam completes almost an additional revolution.

The elimination of two out of three microstructure pulses can also be accomplished in another manner. Use of a beamstop azimuthally far away from the target together with a pulse width smaller than 30 ns at the deflector II permits only every third microstructure pulse to be deflected towards the neutron producing target while the following two microstructure pulses strike the beamstop at maximum radius. Detailed measurements of additional background have not yet been made, but no appreciable amount should be expected, since neutrons are shielded from the flight path by the magnet yoke and the collimators.

The Neutron Time-of-Flight Apparatus. The overall layout of the spectrometer is shown in fig. 2. A thick natural uranium target is used for neutron production. There are two collimators along the flight path defining a narrow neutron beam with an angular spread of 0.5° . The neutrons are detected by a liquid proton recoil detector at the end of an evacuated flight path. The characteristic features of the spectrometer are summarized in table II.

TABLE II TIME-OF-FLIGHT APPARATUS

Flight path	56 - 57 m
Deflection radius	0,930 m (40,5 MeV deuterons) to 1,030 m (50 MeV deuterons)
Time resolution	$(1 \pm 0,3)$ ns full width half maximum
Energy resolution	200 eV at 0,5 MeV
Resolution of Spectrometer	0,02 nsec/m
Integrated neutron flux at $3 \mu A$ target current	$(5 \pm 2) \cdot 10^4$ neutrons $cm^{-2} sec^{-1}$ above 250 keV at 56 m

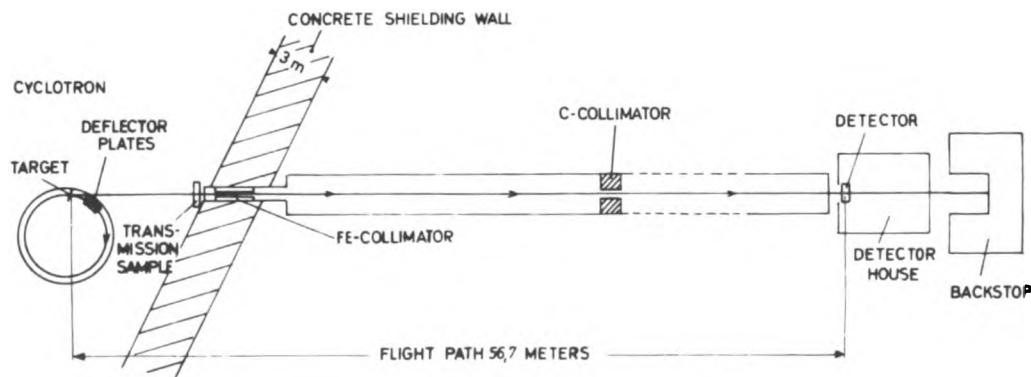


Fig. 2 Geometry of the neutron time-of-flight spectrometer used with the Karlsruhe isochronous cyclotron (top view).

A typical neutron time-of-flight spectrum is shown in fig. 3a. The spectrum is not corrected for the energy-dependent detector efficiency. The maximum at about 20 Mev is due to neutrons from deuteron break-up and/or stripping reactions. The flat distribution at energies smaller than ~ 6 Mev represents the distribution of neutrons from evaporation and fission processes. Use of pulse shape discrimination gives a reduction of the background due to γ -radiation of a factor of about 10.

The energy resolution of the spectrometer was determined by measuring the resonances of Fe^{56} near 512 and 530 kev, respectively. For each energy there are two closely spaced resonances as Beard⁵⁾ has shown. The two resonances at about 512 kev were well resolved in our first measurements. These resonances are shown in fig. 3b.

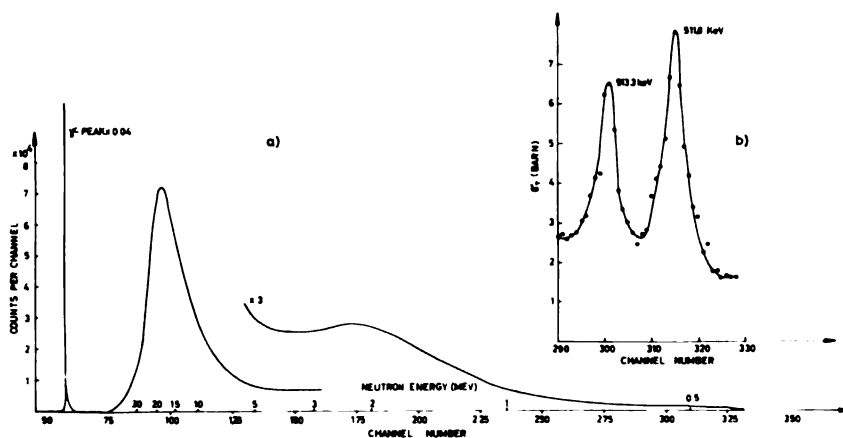


Fig.3 a) Typical neutron time-of-flight spectrum
b) σ_0 for natural iron in the 510 keV region.

From a calculation of the maximum resonance cross section at this energy a resolution of about 200 ev was deduced. The theoretical value at that energy is 165 ev. As an example of operation, the total cross section measurement for calcium in the energy region between 1.4 and 4 Mev is shown in fig. 4.

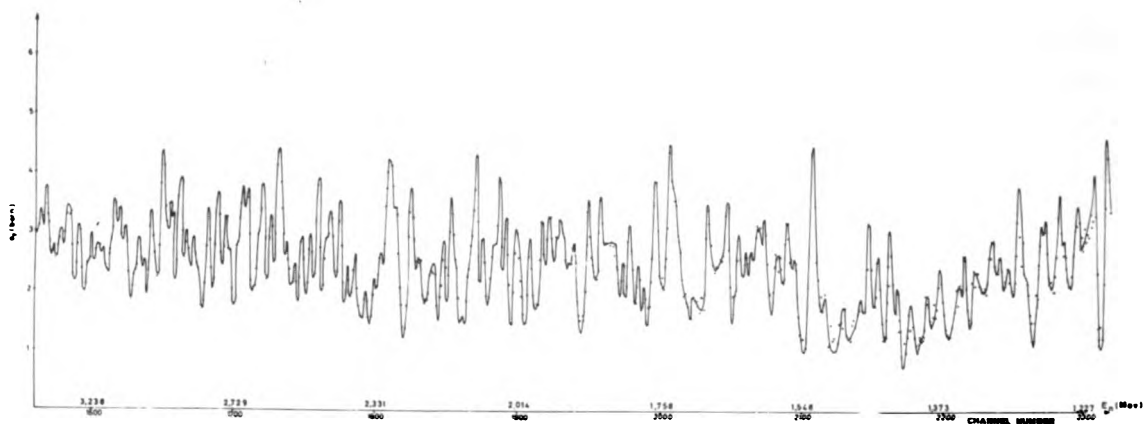


Fig.4 σ_T for natural calcium

sample thickness = 0,113 atoms/barn. Preliminary data.

Possible Improvements: With respect to neutron production aspects the Karlsruhe neutron time-of-flight facility is not yet operating under optimum conditions. The following improvements are planned for the coming year.

First, we plan to increase the repetition rate from 20 kc/s to about 160 kc/s. This leads to a gain in neutron intensity by a factor of 8. Higher power tubes are now available which might prove reliable enough to warrant a basic modification of the H.V. fast pulse generator.

With the present length of the flight path, repetition rates up to 200 kc/s are acceptable if measurements are restricted to neutron energies above 500 kev. For unmoderated neutron spectra from natural uranium targets the restriction to a neutron threshold of several hundred kev seems to be reasonable since the intensity per channel decreases rapidly and background difficulties arise at lower energies. Second, attempts will be made increasing beam current during the pulses. Test runs with more intense ion sources and inclusion of source pulsing are in progress. With increased source strength, it is reasonable to consider increasing the flight path. A 150 m flight path is under discussion. This would give an improvement in energy resolution of a factor of about 3, which is especially important in the energy range above 5 Mev.

3. General Aspects of the Use of Sector-Focused Cyclotrons

Neutron intensities: The availability of high energy heavy particle accelerators for neutron production is certainly not new, but it seems worthwhile to point out here some special characteristic features. Although studies of sector-focused cyclotrons for energies up to 1 Bev ²⁾ are in

progress most of the machines under consideration are designed for lower energies. Therefore discussions in this paper are restricted to particle energies up to 150 Mev.

For a proton of 150 Mev the mean free path for a nuclear encounter in a Pb target is of the order of the ionisation stopping range. So a considerable fraction of the protons strike a target nucleus while protons still have an appreciable fraction of their initial energy. After cascade processes in which single nucleons are knocked out highly excited nuclei will remain. A heavy nucleus excited to about 50 Mev is expected to boil off several evaporation neutrons with an average energy of some Mev. On average, one proton will generate about 1 - 2 neutrons ⁶⁾. Some 20 - 30 % will have energies above 15 Mev, but most of the neutrons arise from evaporation. This evaporation spectrum is then to a first approximation not different from a fission spectrum. As can be seen from table I, sector-focused cyclotrons are operated with average beam currents up to 1 ma. A Pb target of appropriate dimensions, will stop 150 Mev protons, and thereby a line source of 10^{16} neutrons/s is made ⁶⁾. Of course, the proposed Los Alamos high current proton linear accelerator, LAMPF, or the proposed S.O.C. are expected to provide 10^{17} or 10^{19} n/s ⁷⁾⁸⁾. However, with the present principle of beam deflection, it seems possible to concentrate almost the whole average beam current into neutron producing ion bunches of several nanoseconds. On the basis of 1 ns neutron pulse width achieved with the Karlsruhe isochronous cyclotron, burst intensities of about $5 \cdot 10^{19}$ neutrons/s during the pulses should be available with a 150 Mev proton machine with 1 ma average beam current.* The fascinating aspect of the use of sector focused cyclotrons appears from the combination of its characteristics, that is from the relatively high total 4π neutron output combined with small pulse widths and the possibility of reducing pulse recurrence frequencies to several hundred kc/s. Sector-focused cyclotron neutron spectrometers should therefore greatly improve the present capabilities of carrying out high resolution fast neutron time-of-flight experiments.

Fast neutron experiments suffering from a lack of intensity: In the fast neutron region, say from 1 - 100 Mev, high resolution studies on reactions of the type (n,x), in other words (n,p), (n, α), (n,d), (n,T), (n,He³), etc. could be carried out if higher source strengths are available. The object of (n,x) experiments, apart from cross section measurement, is to achieve accurate data about level schemes, level density parameters, and to obtain information about intermediate structure in the Ericson or

* assuming a repetition rate of 200 kc/s

Feshbach sense. Angular distribution measurements additionally provide information about reaction mechanisms, and from the fraction of direct reactions, quantum numbers of nuclear levels and possibly deformation parameters and other properties of nuclei may be obtained. This has been done from some of the available measurements, which have been carried out mainly in the energy region at 14 Mev and in the range between 12 - 20 Mev. For better understanding of nuclear structure, however, more data including also medium weight nuclei and wide energy ranges are necessary as Cindro ⁹⁾ pointed out. The recent development of the idea of intermediate states has also brought a renewed interest in inelastic neutron scattering experiments in the Mev region. Useful information could be obtained from high resolution neutron scattering experiments involving only single reaction channels. In such experiments very intense pulsed neutron sources are desirable.

Rates at which for instance (n,x) measurements are now conducted are very slow and require great patience. With a 500 μ a 200 kv deuteron accelerator with intensities of about 10^9 neutrons/ 4π ·s, a complete angular distribution is obtained within 1 - 3 months ¹⁰⁾, depending upon the cross-section of the nucleus studied and the layout of the experimental apparatus. These times indicate that these measurements are not in a satisfactory state, where experiments should be possible over periods of days.

Of course these experiments also can be done with monoenergetic steady beam sources, but monoenergetic neutron sources involving the usual particle accelerators and reactions with light nuclei cannot cover the whole energy range; for instance difficulties arise in the ranges between some hundred kev and 2 Mev in view of intensities, between 8 and 10 Mev, as well as the energy range above 20 Mev.

Finally it should be mentioned that if very intense fast neutron sources are available experiments could be initiated which would explore the differences that may exist in the nuclear forces for neutrons and protons as Bartholomew, Milton and Vogt ⁸⁾ have pointed out.

4. Conclusions

A comparison of various pulsed neutron devices was given in a recent paper of Bartholomew, Milton and Vogt ⁸⁾. These authors defined a rough figure of merit, viz.:

$$M = \frac{Q \cdot f_{\max}}{\tau^2}$$

where Q is the 4π neutron output per pulse, τ is the pulse width in seconds and f_{\max} is the maximum pulse repetition rate per second. In table III various time-of-flight facilities considered in the above-mentioned

paper are compared with that obtained from our sector-focused cyclotron time-of-flight apparatus. As was discussed at the end of section 2, there are several possibilities for increasing the figure of merit for the spectrometer described. An increase of at least about half an order of magnitude seems to be possible. On the basis of the present figure of merit, which is about $1 \cdot 10^{31} \text{ s}^{-3}$, it is obvious that the facility described represents a powerful tool for fast neutron time-of-flight work.

Table III Figure of Merit for Fast Neutron Time-of-Flight Facilities

Facility	I_a [sec^{-1}]	I_p [sec^{-1}]	f [sec^{-1}]	τ [sec]	M [sec^{-3}]
Rensselaer Electron Linac ^a	$4 \cdot 10^{17}$	$2,6 \cdot 10^{18}$	720	$9 \cdot 10^{-8}$	$2,2 \cdot 10^{28}$
Nevis Cyclotron ^a	$1 \cdot 10^{18}$	$1,2 \cdot 10^{19}$	60	$2 \cdot 10^{-6}$	$3 \cdot 10^{28}$
Pulsed Reactor I.B.R. ^a	$1 \cdot 10^{17}$	$3 \cdot 10^{13}$	83	$3,6 \cdot 10^{-7}$	$2,4 \cdot 10^{28}$
NRU Fast Neutron Chopper ^a	$1,4 \cdot 10^{16}$	$1,5 \cdot 10^{13}$	$1,3 \cdot 10^3$	$8 \cdot 10^{-7}$	$2,3 \cdot 10^{28}$
S.O.C., chopper ^a	$1,8 \cdot 10^{18}$	$1,9 \cdot 10^{16}$	$1,3 \cdot 10^3$	$8 \cdot 10^{-7}$	$3 \cdot 10^{28}$
S.O.C., storage ring ^a	$1,5 \cdot 10^{21}$	$9 \cdot 10^{16}$	200	$3 \cdot 10^{-7}$	$1 \cdot 10^{30}$
S.O.C., microstructure ^a	$5,7 \cdot 10^{17}$	$9 \cdot 10^{14}$	$2 \cdot 10^5$	$8 \cdot 10^{-10}$	$1,4 \cdot 10^{23}$
Sector-Focused Cyclotron actual system	$6 \cdot 10^{17}$	$2 \cdot 10^{13}$	$2 \cdot 10^4$	$1 \cdot 10^{-9}$	$1 \cdot 10^{31}$
Sector-Focused Cyclotron hypothetical system ^b	$8 \cdot 10^{18}$	$1,2 \cdot 10^{15}$	$1 \cdot 10^5$	$1,5 \cdot 10^{-9}$	$5 \cdot 10^{32}$

I_a and I_p are respectively the average and peak unmoderated 4π neutron outputs in the shortest pulses of which the source is capable in neutrons per second.

(a) See reference 8

(b) Assume a 150 MeV. proton beam of 400 μ A int. current, 33% of time av. beam intensity is concentrated into bursts of 15 nsec duration.

Considering generally the availability of sector-focused cyclotrons for fast neutron time-of-flight research the figure of merit might be increased mainly because of the following reasons:

1. It is possible to operate sector-focused cyclotrons at average currents which are about 10 - 20 times higher than those presently used in the Karlsruhe machine (c.f. table II).
2. Machines with higher acceleration energies would show a correspondingly higher neutron output.
3. Using the present principle of beam deflection it seems reasonable to concentrate almost the full average beam current into neutron producing ion bunches of a few nanoseconds.

These facts suggest estimating the performance data for a hypothetical time-of-flight facility which might be used with an advanced sector focused cyclotron. These data are shown in the last line of table III. A figure of merit of about $5 \cdot 10^{32} \text{ s}^{-3}$ appears achievable with this type of apparatus. Of course neutron time-of-flight facilities used with sector focused cyclotrons will involve repetition rates of several hundred kc/s if the high average neutron output will be preserved. Thus the preferred application of such instruments is in fast neutron time-of-flight measurements. But there are many applications for fast neutron work, as was shown in section 3, for which very high neutron intensities are necessary.

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DISCUSSION
OF
PAPERS IV.A. (Havens) and IV.B. (Cierjacks)

Chairman: E. Bretscher

Secretary: B. Diven

WEINBERG: I would say that, from the point of view of the administrator who wants to find out what is going on, the speakers are a little bit too polite to be really terribly useful. What the administrator would really like to know is, for given kinds of experiments, how does one really compare things that are perhaps incommensurable but which have to be compared anyhow. For instance, would the last speaker care to say how his isochronous cyclotron in the 1-MeV region or the 3-MeV region compares with what can be done with the best Van de Graaff accelerator that is commercially available?

CIERJACKS: It is a little difficult to make comparisons with Van de Graaff accelerators, for instance, because there are many techniques which can improve the time resolution. I can compare it with our Van de Graaff in Karlsruhe. What we really get is a resolution of about 0.3 to 0.5 nsec/m with this type of apparatus.

WEINBERG: This means it is a factor of at least 10 less good than what you were stating for your cyclotron--0.02 nsec/m. In your opinion will these devices eventually replace the Van de Graaff for high resolution?

CIERJACKS: No, because the outstanding application with Van de Graaff accelerators is in the lower energy region. For instance, most of the work now is at energies of several hundred keV. Another advantage of Van de Graaff accelerators is that they can use small samples. We have very large beam dimensions because our neutron source is a bit large (about 1 cm by 9 cm). If you work with separated isotopes, you hardly can do it with our machine.

BRETSCHER: As far as I can see you can not do inelastic scattering. This you do much better with an electrostatic generator, and this is rather a big field of importance for reactor physics people generally.

CIERJACKS: Hopefully we can do inelastic scattering work.

BARTHOLOMEW: I feel I ought to say a few words in justification or explanation of the figure of merit that we have used. First of all, I think we were the first to point out that it was a very rough figure in both places that we have talked about and used it, particularly at the Antwerp conference. What we were interested in was

comparing sources in a way that included their versatility, and for this reason we thought the figures should include not only the number of neutrons in the burst but also the frequency of the burst. There is a hope that if you have a high frequency you can reduce it if you want to work at lower frequency, but, if you do not have the high frequency to start, you must live with this limitation. Also we thought the figure of merit should include the pulse width. In principle, you can always widen the pulse width but, if you do not have a narrow pulse, again you must live with this, often serious, limitation. So we used the number QF/t^2 that Havens showed. Now I think if you wanted to use the device in a specific application, or applications, then you should specify figures of merit that apply to those particular applications, and in this respect we would agree with Havens.

BOLLINGER: I have a question or a comment that is related to what Bartholomew just said. Havens' talk stimulated in my mind a new idea for a fast neutron time-of-flight spectrometer. He suggested that the proper criteria to look at, in determining whether you have a good machine or not, is whether the average rate of neutron production is high and whether the number of neutrons per burst is high. This suggests that a very good spectrometer would be, say, an under-moderated reactor where you have a very large beam hole--you open the hole for, say, roughly a day, then close it! This, according to some rough calculations that I just made, will give you an intensity per second that is roughly 10^4 times higher than the Nevis cyclotron and an intensity per burst that is roughly 10^{10} times higher. Does Havens care to comment on whether this a good idea?

HAVENS: What are you going to resolve?

KEEPIN: I think Bollinger's comment on Havens' paper illustrates the ambiguity and the difficulty with selected figures of merit. Havens puts emphasis on two criteria: the maximum intensity in the pulse and time-average number of neutrons from the source. These are indeed two very important criteria, but, even so, a selected figure of merit based on these two criteria alone has seemingly been "shot down" by Bollinger, as one can easily see could be done by citing a rather extreme and, you might say, ridiculous case; nevertheless, I think Bollinger's comment raises an interesting point, which I would like to make. Implicit throughout Havens' talk was the assumption that you have a short enough pulse width consistent with some reasonable flight path for it to be a neutron spectrometer of some reasonable resolution, and under those conditions Havens has covered himself, in his written paper, by specifically saying, "when used as a neutron spectrometer." Then, of course, Bollinger's interesting hypothetical case is excluded. But this is an example of how we physicists can become so oriented to some particular line of experimentation that we tend to forget, perhaps, what constraints or boundary conditions are implicit in our thinking, whereas a differently oriented physicist (e.g., a reactor physicist in the present instance) immediately sees such

implicit constraints. I see this as a source of ambiguity in figures of merit and in approaches to this problem without specific reference to experiments. And, of course, that is why it is good we are all here together to get these different viewpoints on things, but I think what we have just heard does make the Bartholomew figure of merit look just a little more generally applicable than just the two criteria--intensity in the burst and the average number of neutrons per sec.

SCHULT: What was just said illustrates and emphasizes that what we really have to do is not just look at our neutron generators but also to consider the experiments. If we include the experiments somehow in a figure of merit we would automatically exclude all kinds of figures of merit which come up in the way as indicated by Bollinger. But, what Bollinger did is to point out that actually what we have to do is think more of experiments.

MILTON: I would not like to quarrel with Havens' upper limit of 1 GeV as a reasonable number for spallation-evaporation neutron sources, but I think he may have given an incorrect reason for this value. The distance that a proton of energy greater than a few hundred MeV travels through a material is determined not by the range of the proton but by the nonelastic cross section. Above ~ 100 MeV or so, the nonelastic cross section is constant, and, expressed in terms of an interaction length, is about 200 g/cm^2 in heavy elements. In very rough numbers this is ~ 10 cm of uranium or ~ 20 cm of lead, and so only $\sim 0.5\%$ of the protons are going to be transmitted through 60 cm of lead regardless of their energy. Furthermore, this very crude argument would say that the neutron production should be roughly exponential along the length of the target. Indeed, more accurate calculations show that, apart from a slight buildup at the beginning of the target, this is true. The exponential decay rate is just the interaction length of the particle, i.e., ~ 20 cm of lead or 10 cm of uranium. In other words, 60% of the neutrons are produced in the first 10 to 20 cm for any proton above ~ 100 MeV. Therefore, I do not think Havens' argument was correct. The energy that one chooses to use can be justified, or an attempt can be made to justify it, on economic grounds and then it depends on a very large number of assumptions including the number of neutrons that you think you need. Very roughly speaking under some reasonable assumptions the optimum energy is proportional to the square root of the neutron source strength, and it turns out that for 10^{19} n/sec, an energy of one GeV is about an optimum. The cost is rather flat near the optimum but, very roughly speaking, one GeV is a good round number.

HAVENS: I did not give any definite criteria why you should use no more than one GeV; I knew the distributions of neutrons in the target and I took this as a rough rule of thumb, not only for the target consideration, but also for the phase consideration in the acceleration itself. As you go up to higher and higher energy, you usually spread out the pulses in time and therefore they are not as good for neutron

spectroscopy. I was only considering uses for neutron spectroscopy and not for the other applications.

MAIER-LEIBNITZ: My remark is going to be rather general. First, I would say that, if you want to compare instruments, I think one way of doing it is by figures of merit and you can quarrel about how many you will need. I think we would all agree that, once you know the properties of an instrument, you can very easily describe its usefulness for all purposes you have in mind. We have spent quite some time, it would seem, in arguments between users and designers of reactors concerning who should say what he wants and needs and who should say what he can build. That is a problem in itself that should certainly find a solution whenever somebody, who wants to compare two instruments, looks into it and asks the users and considers the properties of the instrument. I do want to mention a second thing which has not come out yet in our discussion. It would seem that, in a given instrument, when you use charged particles there are ways to influence the pulse width, energy distribution in the pulse and the angular distribution which have not been made much use of in the past. The Tandem Van de Graaff, for instance, may be much improved in the future. If you have a certain solid angle at a certain pulse width and if you have a certain energy definition, you can juggle around those three parameters; you can change the energy width while sacrificing pulse length; you can sharpen your pulse while sacrificing energy width; or you can even sacrifice the definition of the area of your beam and gain energy resolution. All such things have to be considered when you compare instruments where you have charged particles because it is amazing what you can do with them.

MICHAUDON: In the calculation of the figure of merit, one considers only the source strength. In fact, what we have to consider is the neutron flux at the detector position. In order to illustrate this, you will probably agree that, for given source strength, the moderated neutron flux from a cyclotron is four times less compared to that of a linac. This is due to the geometrical arrangement.

HAVENS: I do not know about the figure of four. There is certainly some definite advantage to the electron linear accelerator in useful production of neutrons for neutron spectroscopy. I would agree that there are advantages to the geometrical configuration.

SCHULT: How many days per year do you have the use of the Karlsruhe cyclotron to do your experiments?

CIERJACKS: I can not answer this question directly because we have been operating for only about half a year, but we hope we can get three periods over a year with about four weeks per period. We should have such long periods because we have to alter the cyclotron. We must change the flanges on the vacuum chamber and this requires one-half to one day prior to operation.

TUNNICLIFFE: Have you considered the best figure of merit you can achieve in a cyclotron of the type you are talking about when you consider the activation in the machine due to the internal neutron source? This will represent a severe problem from the point of view of maintenance of the machine when the intensity exceeds a certain limit.

CIERJACKS: No, we have not considered this problem.

MARTIN: I would like to comment on Havens' comment about the SOC. We at Oak Ridge do not believe that the SOC is speculative, as it combines well-known principles of other accelerators. I might point out, however, that his use of a storage ring at a 200-kc repetition rate is speculative.

TUNNICLIFFE: I would just like to say I agree with Martin.

BRETSCHER: I have a question for Cierjacks about an experiment: Can you analyze the capture gamma rays from neutron capture in the resonance region? Is your background low enough for such things? This is a very important problem because, as higher angular momenta come in, it will teach us a lot about neutron spectra.

CIERJACKS: It depends on the energy range in which you are working. I think it should be possible.

MICHAUDON: I would like to make another comment on the figure of merit of a neutron spectrometer. What we have been proposing is something like QF/τ^2 , and what Havens mentioned was that it was unfair to take this figure of merit for both slow neutron spectrometers and fast neutron spectrometers because high repetition frequency was not available for slow neutron experiments. But there is another component because τ should be the total time resolution and not only the width of the primary neutron burst of the machine. For instance, at Columbia, they have a 20-nsec burst but, to do an experiment at 100 eV, the moderation time is 200 nsec--10 times more. Therefore, the figure of merit should be dropped by a factor of 100. I think the situation is still worse for the Karlsruhe cyclotron because they have used a value for τ of 1 nsec. Therefore, for the same experiment at 100 eV there is a difference of 200--that means that the figure of merit should be dropped by a factor of 4×10^{-4} , in addition to the repetition frequency for a low energy experiment.

BECKURTS: I think Havens made clear the separation between fast and slow spectrometers, and I think, as far as the Karlsruhe machine is concerned, it is really only a fast neutron spectrometer and you spoil it completely if you moderate it. I really think that these figures of merit should be restricted to fast systems where there is no moderator and we should find some different definitions for the slower systems.

CEULEMANS: If I may just make a short comment, the point that Michaudon made was already explained in Harvey's paper where he included moderation time and instrumental

time resolution.

TASCHEK: May I make one more comment which applies somewhat to Weinberg's request for comparisons. I believe this Seminar has a criterion of very intense neutron sources underlying its discussions, and we have said that is around 10^{15} to 10^{16} n/cm²sec continuous; however, some of these questions of resolution are beginning to get very detailed. They do not apply, for instance, to a Van de Graaff machine. Since with a Van de Graaff machine, you start with a monoenergetic source, you could, indeed, get resolution in the appropriate energy range which is very fine. But the thing that you cannot get is a source of 10^{16} , and that is one reason why you would not stress them here.

SCHULT: I wonder whether one could consider the following possibility: Could one express the figure of merit as a function depending on the neutron energy? What you would get then is a function which has somewhere a maximum. If you then want to do an experiment in a certain range of neutron energy, you find that one device is better there and you can see how the figure of merit decreases and how another device is better.

BRETSCHER: Thank you very much. I hope you will take quickly a piece of paper and put down the energy dependence!

ELECTRON LINEAR ACCELERATORS AS SOURCES OF HIGH INTENSITY NEUTRON BURSTS

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ABSTRACT: Electron linear accelerators are used as nonbooster sources of neutrons in two distinct modes of accelerator operation, in the accelerator steady-state and in the transient or stored-energy mode.

The basic neutron production process in heavy elements is examined and example accelerators proposed for each mode of accelerator operation which illustrate the present state of the art.

INTRODUCTION

The purpose of this report is to give a projection of the potential of the electron linear accelerator as a nonfission source of intense neutron bursts. While some mention of existing capabilities will be made, the emphasis will be upon future capabilities.

In any discussion of the future, one necessarily has to impose some boundary conditions upon his imagination and optimism. In this the general inclination is to take existing technology as a starting point, project a bit further into the future on this base, and to then examine existing major difficulties to see what the future might bring. It is upon this basis that the present report is made.

In attempts to compare the capabilities of the electron linac as a non-fission source of neutrons with other proposed sources, it is important that the above comments be considered. It is tempting to assume the same rf power source capabilities as projected for other programs and to examine the optimum electron linac using this rf source. I will avoid this temptation on the grounds that for really high average flux electron generated sources, where pulse length limits are acceptable, the booster source seems the obvious direction to go and booster sources are the subject of other papers of the conference.

Electron accelerators are presently in use as elements of pulsed neutron sources in two distinct modes, with boosters and without boosters. Booster sources are in operation at Harwell and at Dubna with neutron multiplication factors of about 10 and 100 respectively.

Nonbooster sources are now in use on a number of linear electron accelerators. I would like in this report to examine in a general way what capability now exists for these accelerator systems and what capability might be expected from them in the reasonable future.

As a starting point to the discussion let us ask what are the elements

effecting the limits of any given type of accelerator based neutron source.

These limits seem to resolve around four points:

1. The efficiency for producing neutrons per megawatt of accelerator beam and the characteristics of the source.
2. The feasibility of producing the desired accelerator beam current.
3. The thermal problems in the neutron producing target.
4. The operating costs of the facility including such elements as electrical power, etc.

From an examination of these points it should be possible to define the areas where nonbooster electron linac neutron sources make sense and the eventual capabilities of such sources.

CHARACTERISTICS OF THE NEUTRON SOURCE

The basic characteristics of a neutron source are the neutron yield, the neutron energy spectrum, the neutron source size, and the neutron pulse length. These are discussed below for the unmoderated source.

As has been pointed out previously by Beyster and Russell,¹ unmultiplied neutron yields from various target materials per incident high energy electron are not well known and experimental information is incomplete and contradictory. Figure 1, reproduced partially from reference one, indicates two calculations of the neutron yield expected from a thick U^{238} target as a function of incident electron energy. Experimental measurements of this yield curve are not in agreement and do not exist above 40 MeV. The chief characteristic of the solid curve, developed by MacGregor,² is the continued rise of the neutron yield per unit electron beam power above 40 MeV incident electron energy. An examination of the MacGregor calculation indicates that this rise results from a long high energy tail in the photo-neutron production cross section above the photonuclear giant resonance as indicated by the data of Jones and Terwilliger.³ Confirmation of the existence and magnitude of this high energy tail in the neutron production cross section is shown in Fig. 2, where is shown some recent data by Wyckoff⁴ obtained on the NBS linac. Plotted are integrated cross sections for $Bi^{209}(\gamma, xn)$ for all reactions up to $(\gamma, 9n)$. These data indicate that about 58% of the integrated neutron production cross section results from multiple neutron emission reactions and adds validity to the MacGregor calculation.

Detailed experimental information does not exist on the neutron energy spectrum for high incident electron energies. For lower electron energies the neutron energy spectrum is mostly a statistical boil-off spectrum. For higher incident electron energies a high energy tail in the spectrum would be expected due to direct reactions. For electron energies below 100 MeV, the fraction of the neutrons produced in this high energy tail will be small.

The size of the neutron source is indicated by the calculation of MacGregor.²

For a U^{238} target, 96% of the neutron production takes place in a 3-cm thick target in the direction along the incident beam. The lateral source dimension will be completely dominated by heat removal questions.

ELECTRON LINAC BEAM CURRENT CAPABILITIES

A discussion of the beam current capabilities of electron linacs naturally falls into two categories: linear accelerators operating in the steady-state regime, and linacs operating in the transient or stored-energy regime.

The steady-state behavior of the traveling wave electron linac is a relatively well understood art--although surprises continue to occur from time to time. The basic expressions defining the linac steady-state behavior are given by Eqs. 1 to 4 below for waveguide of uniform structure:

$$V = V_o \cos \phi \left[1 - \frac{1}{2i_m} \right] \quad (1)$$

$$V_o = E_o L \left(\frac{1 - e^{-IL}}{IL} \right) \quad (2)$$

$$i_m = \frac{P_o}{V_o} \cos \phi \left(\frac{\frac{e^{-IL} - 1}{IL}}{1 - \frac{e^{-IL}}{1 - e^{-IL}}} \right) \quad (3)$$

$$\mu_m = 1/2 \cos^2 \phi \left(\frac{\frac{e^{-IL} - 1}{IL}}{1 - \frac{e^{-IL}}{1 - e^{-IL}}} \right) \quad (4)$$

where V_o is the energy gain at vanishing beam current,

V is the energy gain at beam current i ,

E_o is the peak electric field in the waveguide,

I is the rf field attenuation coefficient,

L is the length of waveguide,

i_m is the beam current at which maximum rf power is transferred to the beam,

ϕ is phase angle of the electron bunch on rf wave,

μ_m is the maximum efficiency for transfer of rf power to beam power.

While most modern waveguides are not of uniform structure, these expressions are simpler and suitable for our purposes.

The general behavior of these quantities is shown in Fig. 3 where are plotted V_o , i_m and μ_m versus the attenuation length of the waveguide IL . The important thing to note from Fig. 3 is that for sensible values of IL , the efficiency μ_m for converting rf power into beam power can be as high as 75% or more while still retaining efficient beam energy gain. This efficiency is also independent of the input rf power.

Traveling wave electron linacs of relatively low energy have been demonstrated with steady-state currents of more than 2 A. It is reasonable to project that 5 A is within present capabilities. To give meaning to such an accelerator, consider as an example a 100 MeV, 5 A peak current accelerator. This accelerator would have pulsed beam power of 500 MW peak, and produce a pulsed neutron flux of about 3×10^{18} n/sec. For 75% efficiency of converting rf power, the accelerator would require 670 MW of pulsed rf power or about 35 20-MW klystrons. With presently available klystrons this source could operate with 500 pulses/sec of 2 μ sec duration to produce 500 kW average beam power and about 3×10^{15} n/sec.

The major technical problems in developing the accelerator of the above example would be in assuring the suppression of unwanted waveguide modes which could cause beam blowup and in injector developments. Such an accelerator is however within the state of present day technology. It would certainly be a large facility.

The above example is also useful to illustrate other points. Consider Fig. 1 which indicated two different calculations of the neutron production efficiency for high energy electrons. At 100 MeV the MacGregor calculation is about 30% above the earlier Baldwin, Gaerttner and Yeater calculation. While this seems a rather small amount its importance can be better understood in that it corresponds to about 10 out of the 35 klystrons required for the facility. The importance of better neutron yield data cannot be overemphasized.

The example is also useful to illustrate a crossover between neutron booster sources and nonbooster sources. While the facility of the example is feasible and perhaps not even unreasonable, it is clear that a facility of 10 or 100 times this capability is really large. With accelerator booster combinations however substantial multiplication is possible, while still retaining a 2 μ sec neutron pulse length. To further illustrate this point, the NBS linac, fully powered with 10 MW on each section and a booster having multiplication of 10, would match the neutron production of the above example accelerator.

The more interesting nonbooster electron accelerator source comes when we consider shorter neutron bursts, where the electron beam energy is derived from the stored rf energy in the waveguide. Traveling wave linacs presently exist operating in this mode with beam pulse length from about 50 psec to several tenths of a μ sec and with beam currents up to 2 or 3 A. Accelerators operating in this mode are presently under construction which should produce up to 15 A of peak beam current.

For standing wave linacs the highest beam currents presently achieved are on the PHERMEX facility⁵ where 70 A, 0.2 μ sec pulses have been demonstrated, although at very low duty cycle.

The limitations on pulsed beam current in the stored-energy mode of operation are not well understood. Simple arguments would indicate capability of much

greater beam currents than presently observed. To illustrate this point we consider again a specific example. Assume a linac operating at L band (1300 Mhz) having waveguide sections each 2.5-m long, rf filling time of 1.7 μ sec, and rf field attenuation length (IL) of 0.4. We assume that with 10 MW of peak rf power, P_o , into this accelerator the unloaded energy gain, V_o , for each waveguide section is 20 MeV. This assumed accelerator would thus correspond reasonably well to several L band accelerators now in operation. We will calculate the maximum possible beam current for this accelerator in three different approximations assuming a square-wave beam pulse of current i and length t .

As a zeroth approximation, we ignore the attenuation of rf in the waveguide and assume the pulse length t to be much less than the filling time t_f . Then, for average beam energy of $\bar{V} = V_o/2$, the beam current which totally depletes the energy stored in the waveguide in time t is given by

$$i = 2 \frac{P_o}{V_o} \frac{t_f}{t} \quad (5)$$

$$= \frac{t_f}{t} \text{ A for this example.}$$

Results of this simple calculation are shown in Fig. 4. They indicate for example that for a 5 nsec beam pulse, 340 A pulses are possible.

A better approximation including attenuation and other nondispersive effects can be made by calculating the current i at which the beam energy gain in the section is reduced to zero in time t . Such a calculation yields⁶

$$i = \frac{V_o I}{r_o} \left[\frac{1}{IL \frac{t}{t_f} e^{-IL t/t_f} - (1-IL) (1-e^{-IL t/t_f})} \right] \quad (6)$$

Results of this calculation for the example accelerator are also plotted in Fig. 4. They would seem to confirm the validity of the simple approximation above and indicate a possible current of 230 A in a 5 nsec beam pulse which is reasonable since rf attenuation is now included.

A difficulty of the above calculations is that they assume the beam to be a simple wave at the operating frequency of the accelerator. This clearly is not a good approximation for short pulses. Accelerator waveguides typically operate with passbands of only a few megacycles and additional dispersive effects can be anticipated. A calculation including these dispersive effects indicates a greatly reduced beam current capability for short beam pulses.^{7,8} Results calculated for the above example accelerator are plotted in Fig. 4. They indicate a possible current of 42 A in a 5 nsec beam pulse.

The beam currents calculated in Fig. 4 represent upper limits. For these currents and pulse lengths, the beam energy gain would be reduced to zero at the

end of the beam pulse. A more realistic estimate would be a reduction of beam energy to one half its peak value, in which case the beam currents would be one half of those shown. Thus a short pulse beam current of 15-25 A is the capability of the accelerator of the above example.

It should be commented that the limits set by dispersive effects to short pulse current in traveling-wave linacs have not been tested experimentally. Injection and first acceleration problems in existing linacs have made observation of these effects difficult. Work is proceeding in a number of organizations on improved high current injection systems which should provide this information as well as an increase in short pulse current on existing facilities.

Accelerator theory including dispersive effects has not been carried out to determine the nature of accelerators designed specifically to optimize short pulse peak current. Work should be done to determine the influence of operating frequency and waveguide band pass as well as to study possible standing wave structures.

THERMAL PROBLEMS

The above discussion has ignored the problems of heat removal in the neutron target. It appears clear that an eventual limit will be reached where the attainable neutron flux will be determined by a combination of average heat removal and fatigue failure caused by pulsed heating. For very short pulses, heat removal has not yet become a dominating problem and will not for some time although fatigue failure due to thermal shock is already a problem.

For longer pulses, such as the example accelerator described, which would have 500 kW of average beam power, heat removal would be a serious problem. This is especially true, since because of the electromagnetic shower, the total heat is concentrated in a relatively small volume. In this case one might be forced to a liquid metal target system as has been proposed by Bartholomew.⁹

CONCLUSIONS

Electron linear accelerators are in use in a number of installations as sources of high intensity bursts of neutrons. For sources designed to enhance the neutron production, a 100 MeV electron accelerator seems optimum and should have a neutron production efficiency of 5.5×10^{15} n/MW sec of electron beam power. More experimental information on neutron production rates versus electron energy is needed.

For accelerators operating in the steady-state mode, a 100 MeV, 5 A accelerator is feasible. Such an accelerator could produce 500 2-usec bursts of n/sec having peak intensity of 3×10^{18} n/sec and average intensity of 3×10^{15} n/sec.

Electron linacs operating in the stored-energy mode, with nsec pulsing should produce pulse currents of about 20 A for 5-10 nsec pulses or peak neutron fluxes for a 100 MeV accelerator of about 10^{19} n/sec. This is about 10 times the neutron production available on existing electron linac facilities. Present source limits are

determined by injection and initial acceleration problems; however the ultimate limit of such sources seems to be set by dispersive effects in the accelerator waveguides. These dispersive effects are not well understood and need much more study.

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FIGURE CAPTIONS

- Fig. 1. Neutron production by electrons from a U^{238} target plotted as a function of incident electron beam energy. The dotted curve is from Baldwin, Gaerttner and Yeater. The solid curve is by MacGregor. This figure is taken from Ref. 1.
- Fig. 2. Integrated photon absorption cross section for production of neutrons in Bi^{209} by the (γ, xn) process plotted versus the neutron multiplicity as determined by radioactive daughter product production. These data by Wyckoff⁴ indicate that 58% of the neutron production cross section results from multiple neutron emission processes.
- Fig. 3. Steady-state behavior of traveling wave electron linacs for constant structure accelerators. Plotted are the unloaded energy, V_0 ; the beam current at maximum power transfer to the beam, i_m ; and the efficiency for converting rf power to beam power μ_m , plotted versus the attenuation length of the waveguide section.
- Fig. 4. Maximum possible short pulse beam current for example accelerator plotted versus beam pulse length for different methods of calculation. Curve (a) ignores rf attenuation and assumes pulse length short compared to filling time. The current shown is that necessary to completely deplete the stored energy in the waveguide in time t . Curve (b) represents a proper calculation as long as dispersive effects can be ignored. It represents the beam current which will cause the beam energy gain to equal zero at the end of the pulse. Curve (c) is the same as curve (b) except that dispersive effects due to the narrow pass band of the waveguide are included.
In a practical situation about one half the current indicated by curve (c) might be achieved. In this case the energy spread of the beam would be 50%.

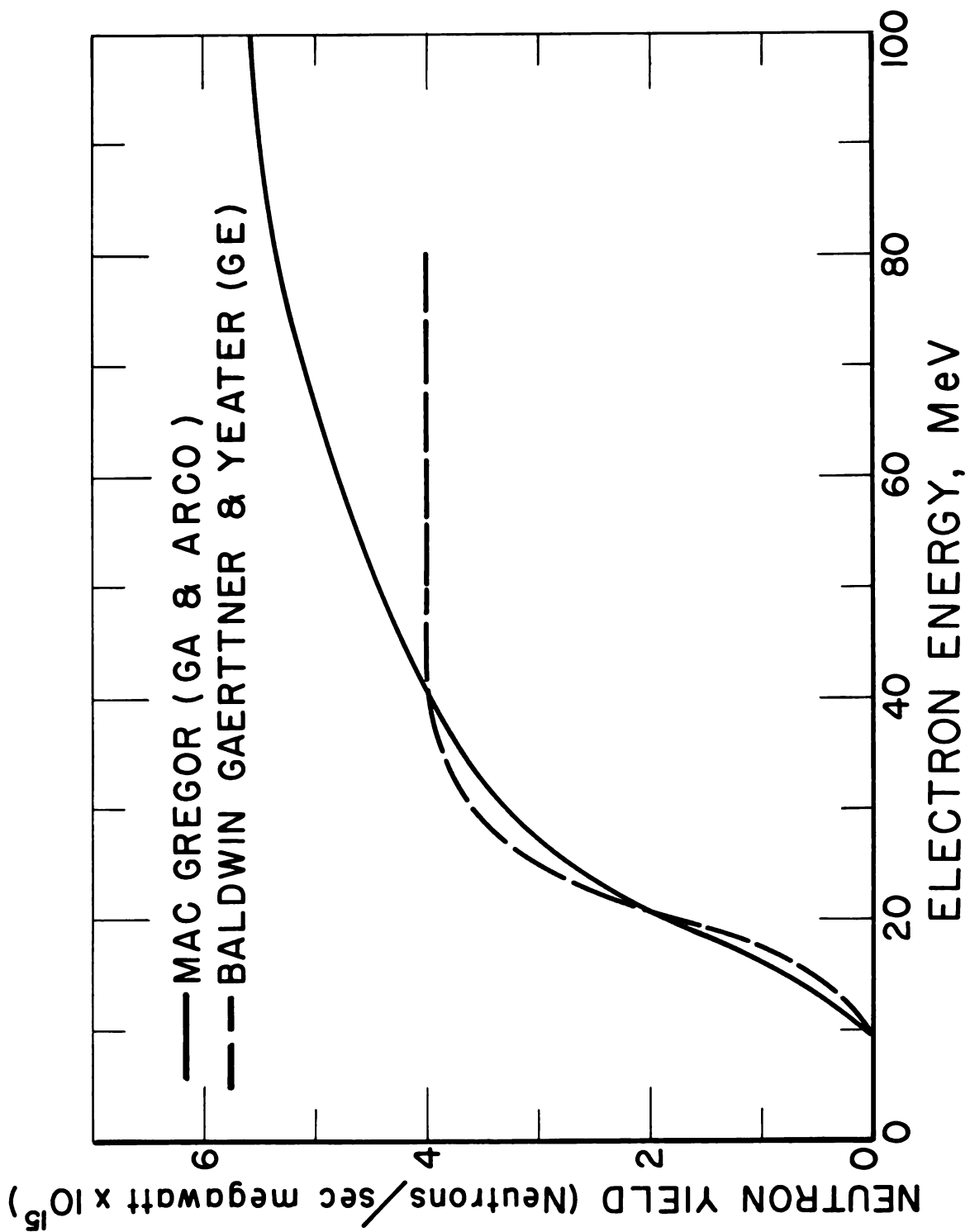


Figure 1

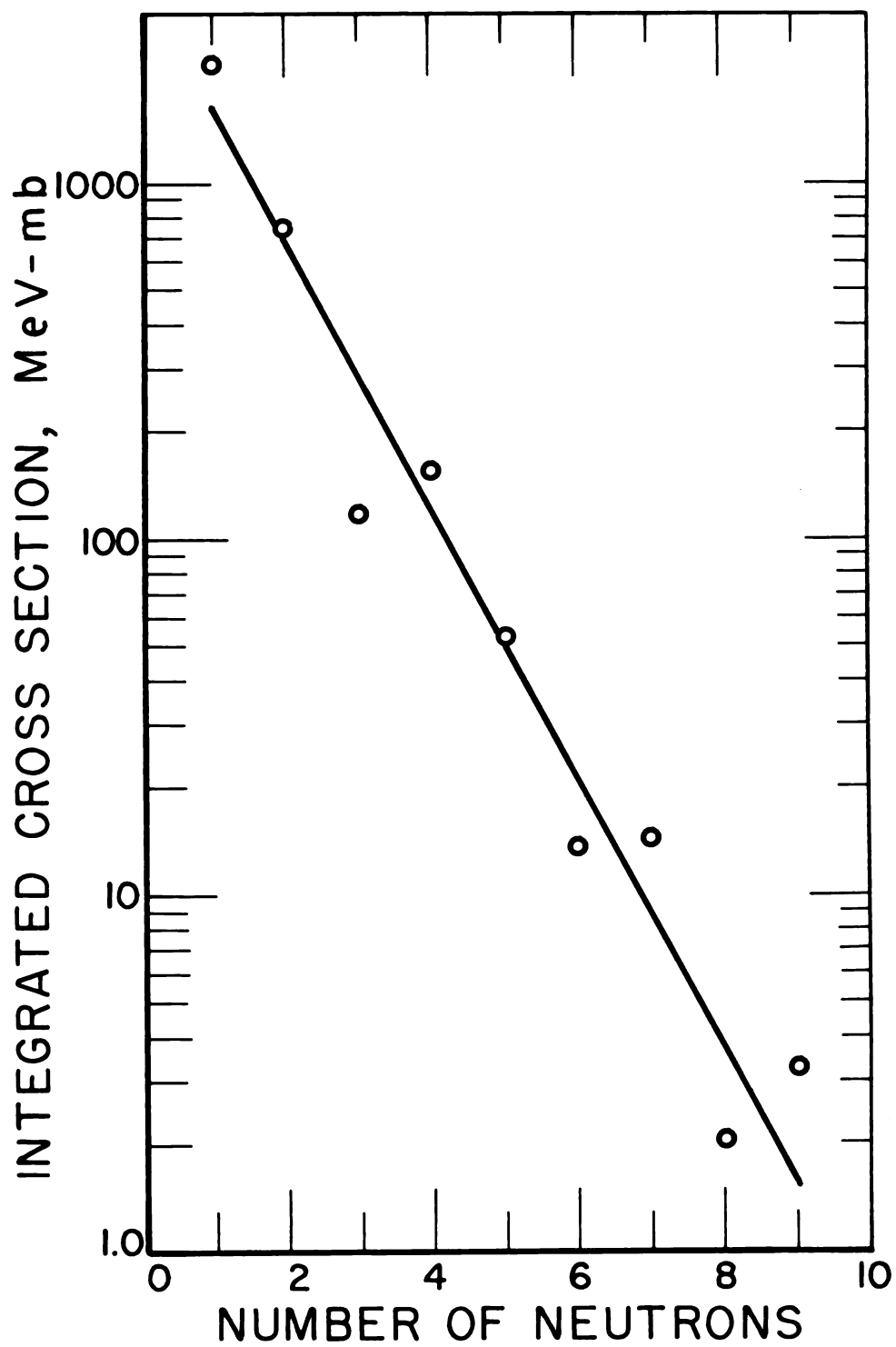


Figure 2

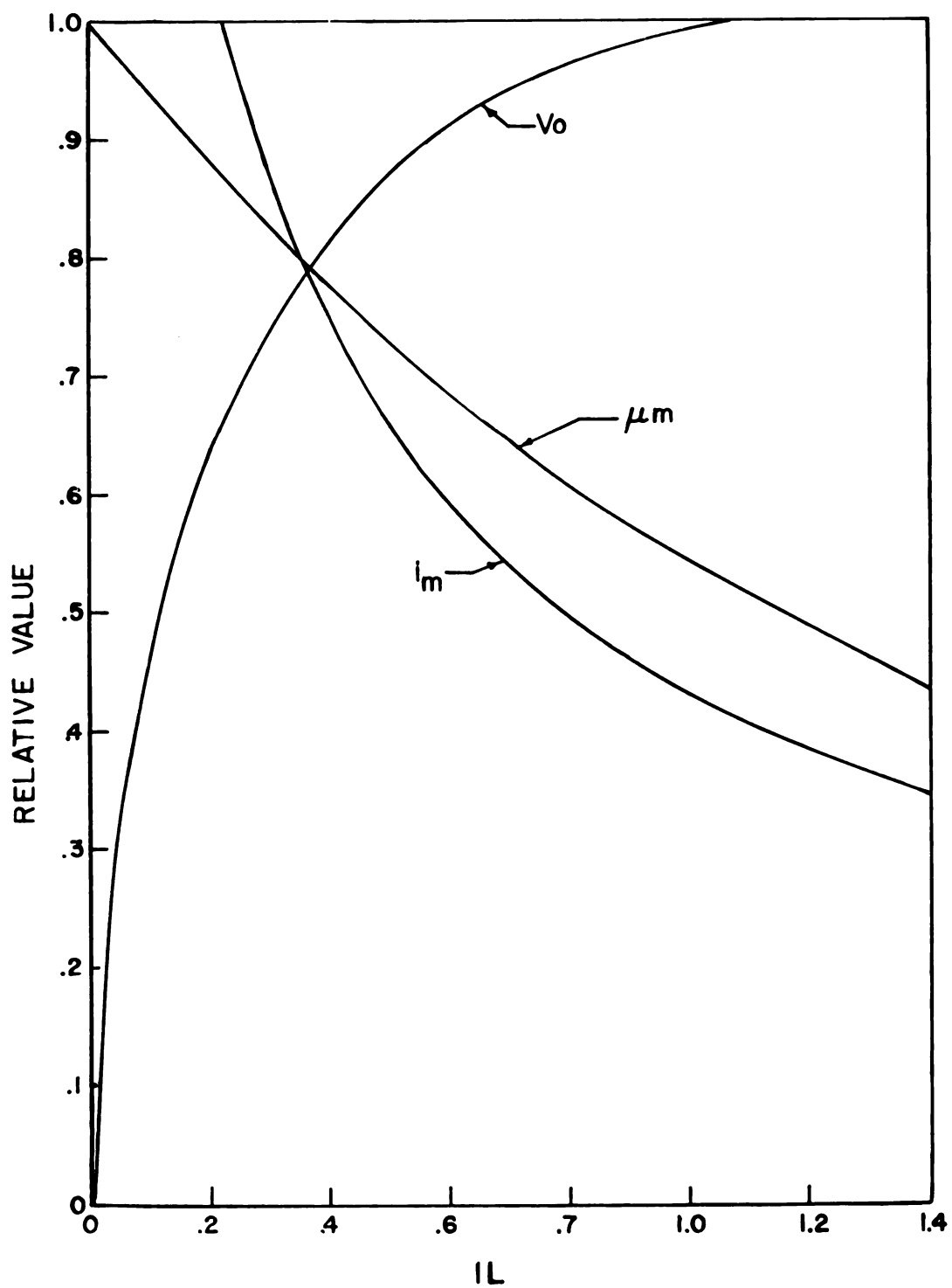


Figure 3

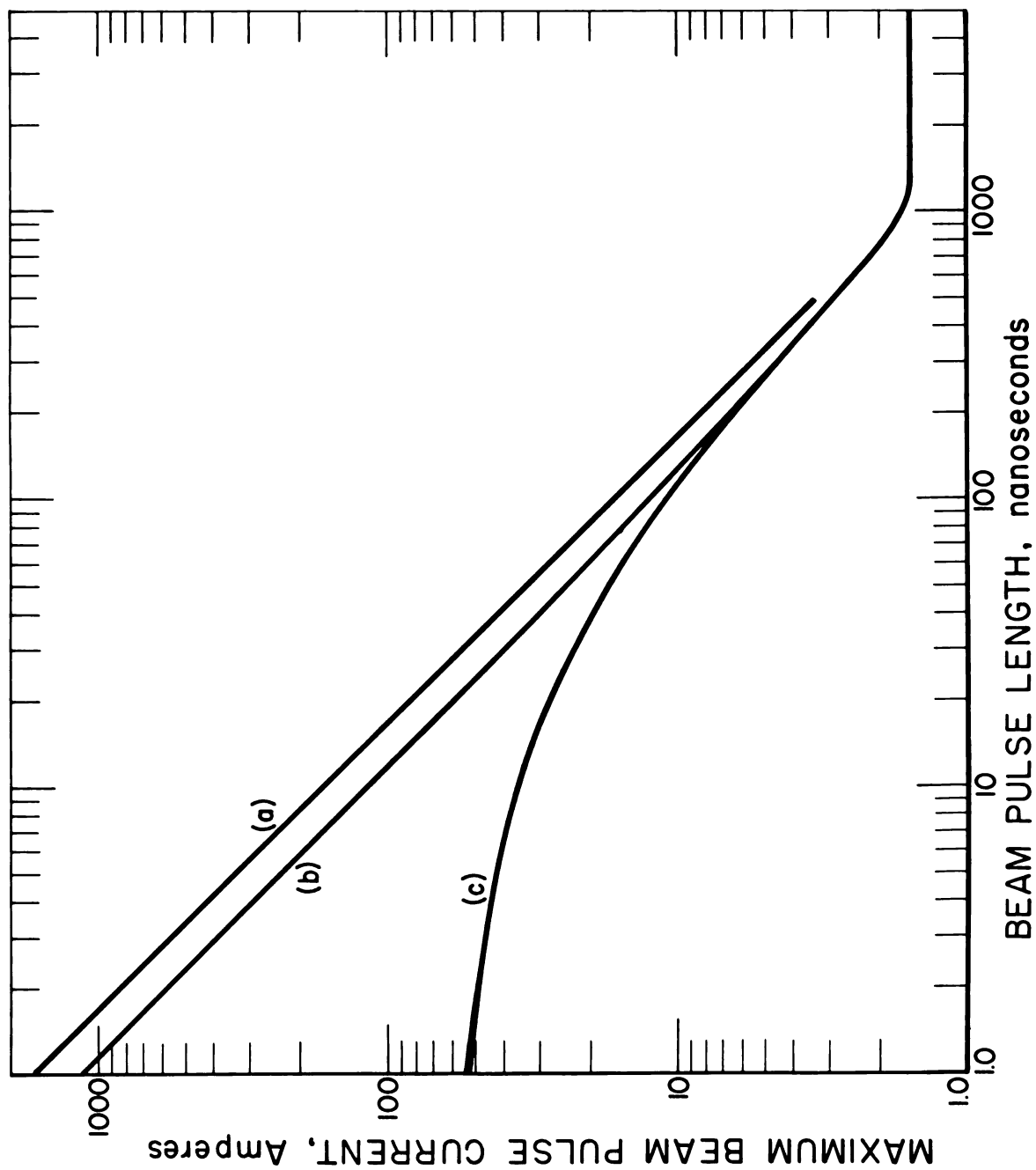


Figure 4

DISCUSSION
OF
PAPER IV.C. (Leiss)

Chairman: E. Bretscher

Secretary: B. Diven

BRETSCHER: If you make a superconducting linac, does it help you?

LEISS: I neglected to comment about the superconducting linac. In a steady state linac, that is, a linac with long pulses, it is very clear that superconductivity does not help you. You are using 75% of your RF power already. For short pulses, it really is not at all clear how much it helps you. It saves you in the microwave power bill enormously, but you buy a large refrigerator. Also, if you want to pulse the accelerator at some reasonable repetition rate, there are some very serious questions about what happens to the energy from the pulse 1 msec earlier, and 2 msec earlier, and 10 msec earlier. If you are truly superconducting, that energy is still in the system. Making the first superconducting linac is a problem and knowing what it will do is beyond the present state of the art.

LARRIMORE: In regard to using these linacs with boosters, I am in a state of ignorance about the accelerators with very short pulses that you mentioned. For a booster you can see it is very interesting to have accelerator pulse lengths of something like $1/2$ to 1 μ sec. There is no reason to go much below this, because most of the work can be done with these pulse lengths. The minimum pulse width of the booster would be something like 5 to 7 μ sec, so there is no reason to go below about $1/2$ μ sec pulse width but there are reasons not to go very far above this, because the larger you make the accelerator injection on time, the smaller the amplification you can use in the booster and the larger the accelerator power must be. Would you say something about the use of short-pulse machines in this range or the comparisons? Suppose you want an accelerator on-time of, say between $1/2$ and 1 μ sec, how would you compare these two machines?

LEISS: This is a hard question to answer because it gets involved in the details -- at what frequency, your filling time and so forth. What we have observed with the NBS linac, which has a filling time of 1.7 μ sec, is that it has a steady-state current of $1/2$ A. We find that, if we go down to $1/4$ - μ sec pulse width, we can get something like 2-A beam current from essentially $1/4$ μ sec and shorter. The current in these shorter pulses is injector limited, however. At some point the current is not injector limited, and I would not quite care to say what that is. But certainly, when you are getting

off the tail of the transient, you still have stored energy advantages. You would have to take a particular accelerator and answer the question. You can get more than the steady state numbers, I would say, factors of 2, 3, 4--numbers like that.

NAGLE: Don't you observe pulse shortening at these higher beam currents?

LEISS: Do you mean at several amperes or at one-half ampere? We observed pulse shortening before a half ampere if our machine is lined up to within 10/1000 in., but, if we misalign it the way it was installed, with every sector off by 1 mm, we don't see it. But it makes energy variation very hard, so you have to start asking what the experiment requires; we do not have many experiments that require 100 kW of beam right now, so we left the machine well lined up. At the higher currents it will come back to plague you -- it is clear that to get 5 A out of the machine there are two areas you have to worry about. One is the pulse-shortening problem, and the second is the injector problem. There is work going on in injectors that should produce something like 30 to 40 A now, and maybe in the near future we will have some. To avoid pulse shortening, I believe that by proper design of the accelerator -- adjusting the parameters of different sections so that the frequencies of the pulse-shortening mode come where they ought to and are not all at the same place -- one can build an accelerator now that will give 5-A steady state. I believe that is the state of the art. Two A in 10 μ sec pulses have been observed in the Boeing machine.

MAIENSCHNEIN: In comparing the neutron yield vs energy calculated by McGregor and the measurements of Baldwin, is it necessary to take into account the thickness of the target? I think the targets were thicker for the calculation than they were in the measurements.

LEISS: It is certainly true that you have to consider target thickness. Actually, McGregor calculated it for a number of target thicknesses: 2, 3, 4, 5, and maybe even 1 cm. The comment I made during the talk was that if you have a 3-cm target thickness you get something like 96% of the maximum neutron production that is possible. With a 5-cm target you obtain something like a 10% multiplication by various neutron interactions. I am not familiar with the target thicknesses that Baldwin used. I might comment on the extreme importance of getting good information on this. In the steady-state accelerator that I described with 35 klystrons, which is a big machine, 30% difference in neutron production efficiency may not sound like much, but it is equivalent to 10 klystrons. Thought of in these terms it is a great big, important, expensive thing to the man who builds accelerators, and it ought to be measured and known. Efficiency bears a very large dollar sign in the facilities if you play the numbers game one way, or it bears a substantial effect in the number of neutrons you will have if you want to play the numbers game a different way. Some of this basic information is very important and it is dependent on geometry, etc.

TUNNICLIFFE: I recall that the conversion of particle kinetic energy to neutrons is much more efficient for protons using the spallation reaction than from the (γ, xn) reaction. I think it would not be unreasonable for the people, who are considering accelerator-driven boosters or accelerator-pulsed reactors, to look seriously into the question of a proton accelerator, perhaps like the Karlsruhe machine. The overall efficiency from electrical input energy per neutron delivered to the booster might be very competitive with an electron linac at the powers they are talking about. I am not necessarily saying the proton linac is the best machine; it might be one of the synchrocyclotrons, or it may be an AGS -- I just have not put in the numbers; I do not know what the answer is, but I think it bears examining.

LEISS: The comment is very fair. If you take McGregor's calculations you are talking something like 1000 to 1200 MeV per neutron, whereas if you take the BeV protons you have something like 50 MeV per neutron, and so you have a factor of 20 to play with. However, you have to take into account various factors that are against the proton, the size of the target, the energy spectrum of the neutrons, which is really quite high and would substantially stretch the boosted pulse length. I would not be a bit surprised if the proton accelerator proved best, but I think it is not at all obvious, even with a factor of 20 to play with.

MICHAUDON: I agree with you about the target and would like to compare the more efficient way to produce neutrons with electron and proton machines. I would like to mention another factor which is the size of the target. With a proton machine, not only is the energy heat per useful neutron lower, but it is dissipated in a large volume instead of a few cc for an electron machine. Therefore, the heat removal for a given beam power is easier with a proton machine.

LEISS: I think the heat removal per neutron is easier because of this factor of 20 between the efficiencies. However, as was pointed out yesterday, you really do not lose many of the neutrons if you spread the target out with low Z materials, such as cooling media. Its radiation lengths that we talk about and the radiation lengths of uranium or lead are so vastly different than that of water or sodium that they really do not count. You do not lose neutrons, and I will spread it out as you please. The heat removal problem is only the difference in efficiency of producing neutrons. Clearly what you want to do is keep the target as small as you can technically handle.

MICHAUDON: I am speaking about the heat removal of the same power into two different targets -- one for electrons, the other for protons. Don't you think it is easier to remove the heat produced with strong protons because it is dissipated in a larger volume?

LEISS: What I was trying to point out is that it is not obvious, it depends on how you design your target. I have a radial dimension to play with; I can spread my target out in thin foils with water in between. I don't think this is at all clear.

MICHAUDON: Your statement is that the same amount of heat can be removed as easily with an electron machine as with a proton machine.

LEISS: Sure, within reasonable bounds.

BARTHOLOMEW: I had a comment with respect to the one of Tunnicliffe -- I think he was referring to a Karlsruhe machine and not a 1-GeV accelerator to be used with the booster. I think the efficiencies compared with the numbers that you quote, which were 5.5×10^{15} n/MW/sec, and I think for the Karlsruhe machine 8×10^{16} or a little more than a factor of 10 improved efficiency, and for the 1 GeV is 3×10^{17} n/MW/sec for the kind of target that we talk about.

LEISS: I am willing to be corrected if I am wrong, but I think it is still true that the photon interaction tends to produce neutrons in a moderately low energy boil-off spectrum far more than the proton does. Even at the Karlsruhe machine energy, more of your neutrons will be at higher energy, and I do not see how this could fail but have adverse influence on the booster.

BRETSCHER: Could I ask you a question concerning the possibility of dual linacs pulsing opposite sides of the target? In Harwell we have a linac and we program the beam with the help of a flipping magnet. If the deflecting magnet is not energized, the beam goes straight; then if we use the magnet, the beam goes to the right or to the left. Now if you use this system here, you use only one linac and then deflect the beam.

LEISS: I would rather buy two injectors -- it is cheaper than the pulsed magnet.

BRETSCHER: We are great enthusiasts about the magnet but it deflects only to a few degrees.

LEISS: Remember that we are doing it at 100 MeV at high repetition rates, plus the fact that the linac is actually easier to build. If we say we are going to talk about one linac, I can put half the current in it and make my technical problems easier. I am sure it is cheaper to use two machines. You can stack them one above the other; the building cost is quite unchanged.

COLE: We have had a number of discussions on how reliable reactors are and, also, on how expensive high flux reactors are to operate. We have had at least assertions that nuclear explosives are inexpensive in terms of cost per neutron, and I presume that they have reached some degree of reliability. I wonder if you would like to comment on the operating cost of a machine such as you described, in terms of neutrons delivered, etc. I think it is not too hard to get the electrical power input, but how about reliability, on-time and maintenance problems, let us say, with the klystron system, etc?

LEISS: That is a loaded question, as you may all appreciate. I do not think there

is any doubt that the standard high flux reactor is vastly more reliable than the linear accelerator of any sort. The pulsed booster is being extrapolated from 6 kW to 10 MW which may be all technically understood, but it is still a substantial extrapolation. Whether that will have the kinds of maintenance problems that a linac has, or a normal reactor has, is not at all clear, at least not to me. There are many horrible things I can think of that could occur in these advanced pulsed boosters. I can not give you a good number for the linac maintenance cost. Let me comment on the reliability, however. If you consider, say, this machine with 35 klystrons, and you would put, say 4 klystrons or 10 or 12% on standby (which would not be a very large escalation of the cost of the machine), I think the experience that SLAC is finding is that with that kind of number on standby you can have your on-time up around 90%. Clearly, the reliability is a problem. With Carl Muehlhause at the Bureau of Standards, we have compared fuel costs against klystron costs, and actually it surprised me that the price of running our linac and the price of running this reactor are not going to be very different. Fission fuels and klystron costs come out very much the same.

AN INTENSE SOURCE OF NEUTRONS FROM THE DENSE PLASMA FOCUS*

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ABSTRACT: The dense plasma focus (created in a coaxial accelerator) is an intense source of neutrons. The plasma focus can be described as a fast dynamic z-pinch in which stored magnetic energy is rapidly converted to plasma energy and then compressed by its self magnetic field. Plasma pressures (nkT) of the order of 3×10^5 atmospheres are developed on axis in a region of ~ 0.5 -mm diameter and 3-5 mm in length. The deuterium plasma density $n \simeq 2-3 \times 10^{19}/\text{cc}$, electron temperature $T \sim 5$ keV and a neutron pulse duration of ~ 0.15 μsec lead to a large neutron emission which is not inconsistent with the assumption of thermonuclear origin.

The performance of the plasma focus in terms of neutron production, quoting only what has been achieved is $\approx 2.2 \times 10^{17}$ neutrons per sec. [$D + D \rightarrow \text{He}^3 + n$ (2.45 MeV)] in a burst time of ~ 0.15 μsec for a condenser energy of ~ 40 kJ. Scaling studies from 10 - 40 kJ of condenser energy result in an almost linear increase of neutron production. The substitution of a 50/50 gas mixture of deuterium and tritium [$D + T \rightarrow \text{He}^4 + n$ (14 MeV)] for the pure deuterium gas shows an increase in neutron production by a factor of 80 - 100. D-D and D-T neutron flight time measurements were made over a flight path of 12 meters; their velocities correspond to neutron energies of 2.45 MeV and 14 MeV respectively.

Scaling of the dense plasma focus to larger input energies and larger physical size should permit on the basis of experimental evidence higher neutron production. For example, condenser energies in the 250 kJ range should yield a neutron production rate of $\approx 10^{18}$ n/sec from the D-D reaction and correspondingly higher ($\approx 10^{20}$ n/sec) for the D-T reaction.

The characteristics of this type of source which are relevant to its potential use as a neutron source in experiments will be discussed.

1. INTRODUCTION

The dense plasma focus device¹⁻⁴ is an intense source of monoenergetic neutrons. The device takes the general form of a hydromagnetic coaxial accelerator in which the hot dense deuterium plasma is created by the rapid radial collapse or pinching

* Work performed under the auspices of the U. S. Atomic Energy Commission and Defense Atomic Support Agency.

action of a high current axisymmetric current sheath toward the axis as the sheath leaves the end of the center electrode. This type of mechanism can be described simply as a z-pinch effect and is known already to be a powerful method of heating and compressing a gas to high temperatures and densities respectively. The neutron production rate per unit volume from a thermal deuterium plasma depends on n^2 (n = particle density) and very strongly on the plasma ion temperature T_i ; for T_i in the range of 2-10 keV, the neutron rate varies approximately as T_i^4 . In this sense, any partially ionized high Z contaminant immersed in the plasma will inhibit the attainment of high temperatures and will reduce, through radiation loss, the final plasma energy density.

2. DESCRIPTION OF APPARATUS

The schematic of the dense plasma focus device is shown in Fig. 1. It consists basically of two coaxial cylindrical copper electrodes (inner electrode 5 cm diam., outer electrode 10 cm diam., and approximately 20 cm long) which fit to a low-inductance header. The pyrex or high alumina ceramic insulator projects along the center electrode a distance of ~ 5 cm. This insulator geometry predicts essentially the shape of the current sheath at early times--the axisymmetric current sheath shape within the annulus for all intensive purposes is shaped like a "bullet". The center electrode is usually at a positive potential. The entire coaxial geometry fits within a pyrex vacuum vessel.

The condenser bank consists of 90 μF at 20 kV and, under shorted conditions at the header, is capable of delivering $\sim 1.3 \times 10^6$ amperes peak current with a current rate $di/dt \sim 1.2 \times 10^{12}$ amperes/sec. The condenser bank is switched by 6 vacuum spark gaps via 36 coaxial low-inductance (25 nH/ft) cables. The total external inductance and resistance is ~ 16 nH and 2.9×10^{-3} ohms respectively. Current is measured by a shielded current loop and voltage by a fast resistance divider network.

Pressure measurements are made with a Wallace-Tiernan pressure gage and autovac gage. The static filling pressures in the discharge section range from 1.5 to 6 torr of deuterium--the optimum pressure depending on the electrode length and applied voltage.

Total neutrons are detected by the silver activation method with four Geiger Mueller tubes embedded in paraffin--a fast scintillator photomultiplier circuit records the pulse shape. Small magnetic probes, double floating electric probes and coaxial electric probes measure the B_θ magnetic field, electric field E_r and voltage drop IR across the plasma sheath respectively.

3. OPERATIONAL SEQUENCE OF DISCHARGE

The vacuum vessel is filled initially to a high pressure of deuterium. With no preionization the applied voltage produces breakdown in the region near the pyrex insulator. Current flows from the center electrode back along the insulator to the

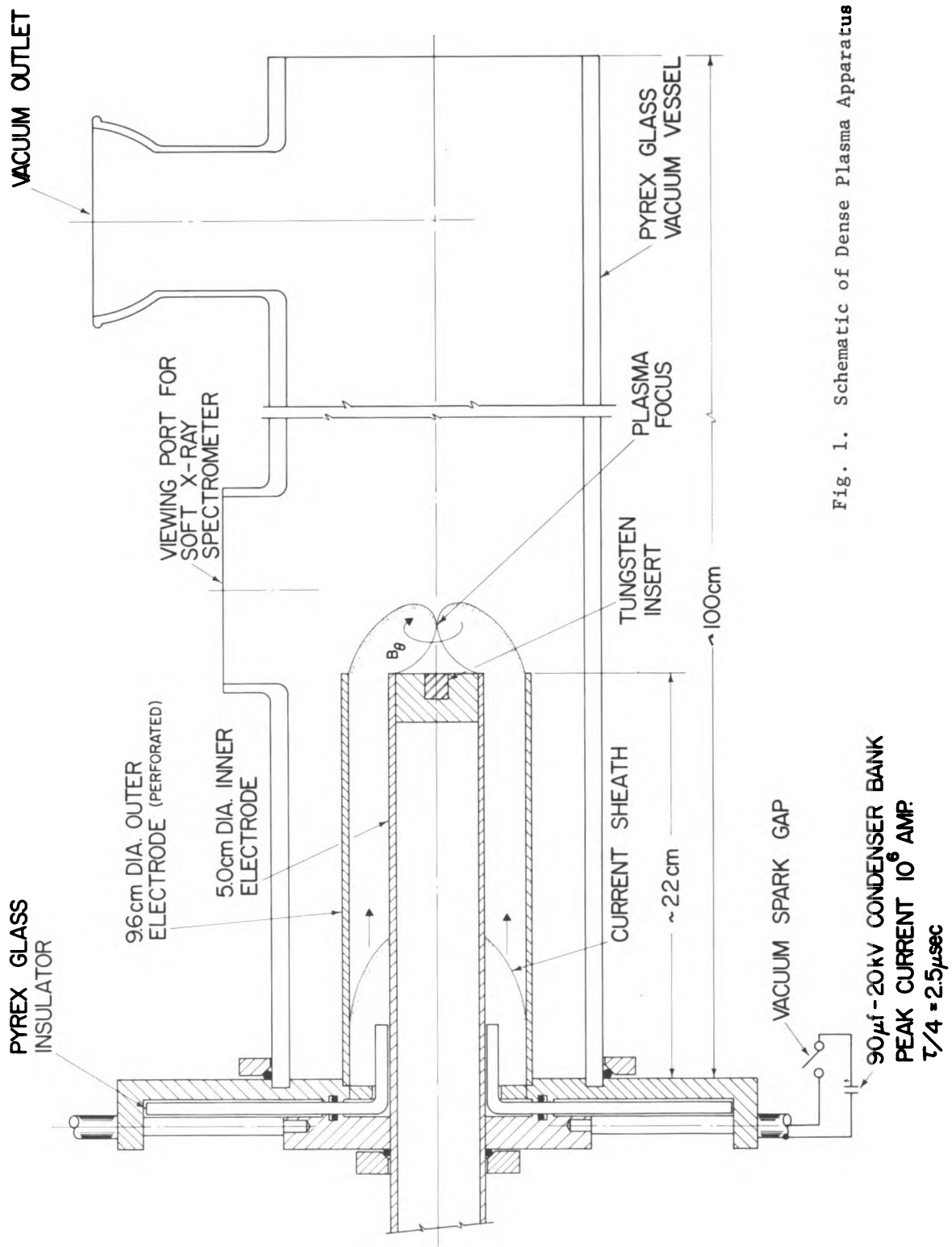


Fig. 1. Schematic of Dense Plasma Apparatus

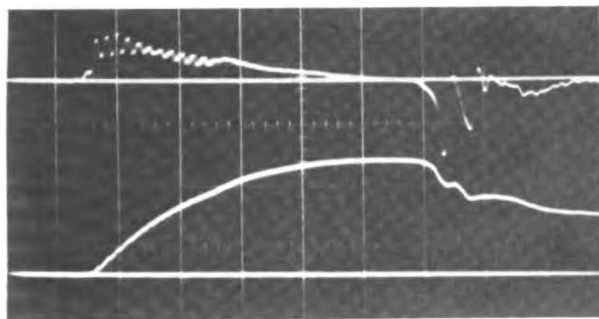
back plate, simulating a "water-fall" flow over the insulator surface. A separate study⁵ has revealed that during the early stages of current buildup, a complex filamentary light pattern is observed at the back plate and that this structure of the light pattern retains its identity as the $j_z \times B_\theta$ inverse pinch force drives the current sheath outward toward the inner surface of the outer electrode. After reaching the outer electrode, the complex filamentary pattern blends into a axisymmetric uniform discharge and the current sheath is then driven axially by the force $j_r \times B_\theta$. The velocity of the current front is larger near the inner electrode surface than at the outer electrode surface--this is expected because of the radial variation of the axial force $j_r \times B_\theta$ with radius. This effect, in addition to the insulator surrounding the center electrode, produces a canted current front which leads at the inner and lags at the outer electrode surfaces. Suffice it to say, the sheath is driven axially by the $j_r \times B_\theta$ force snowplowing a major portion of the gas it encounters. For the optimum conditions of electrode length, voltage and filling pressure, the sheath arrives at the end of the center electrode at the time of peak current. At this time, the axial force $j_r \times B_\theta$ and the sheath mass density is the largest and hence the sheath continues off the end of the center electrode. When this occurs, the axisymmetric current sheath collapses rapidly inward the axis as a result of the strong radial pinch force $j_z \times B_\theta$.

4. NEUTRON SOURCE

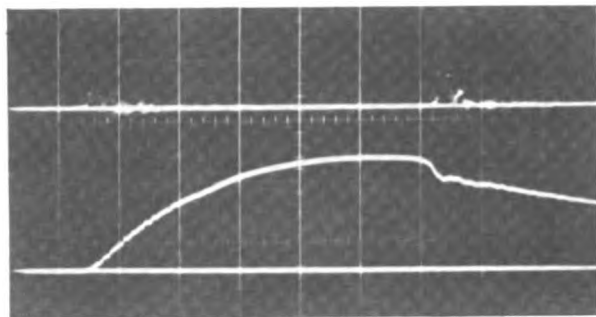
The characteristics of the neutron source are (1) small geometrical size, diameter ~ 1 mm, with a volume $\sim 3\text{-}5$ mm³, (2) the source is positioned $\sim 1.5\text{-}2$ cm beyond the face of the center electrode, (3) the neutron distribution is almost isotropic--an anisotropy of $< 5\%$ has been measured and (4) the full width half maximum neutron pulse width is $\sim 0.1\text{-}0.15$ μsec . A typical neutron pulse and discharge current waveform (bottom trace) is shown in Fig. 2-5 (sweep speed 0.36 $\mu\text{sec/cm}$) for a neutron production of $\sim 9.2 \times 10^9$ neutrons. In Fig. 2-1 and 2-2 is shown the di/dt and voltage behavior respectively along with the current waveform. The voltage across the discharge and the peak current at the time of collapse is ~ 20 kV and 800 kA respectively.

The closest distance of approach to the plasma focus has not actually been determined; however, experiment shows that a metal plate can be positioned within a few centimeters of the plasma focus without seriously affecting neutron production. For higher energy systems, the region beyond the center electrode will become extremely violent and it is expected that the closest approach will be determined by both the strength of materials and the plasma contamination from the electrodes.

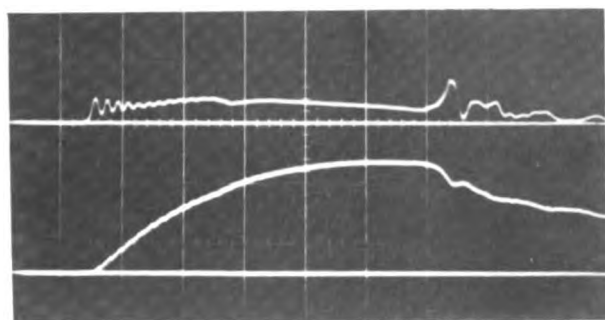
The basic apparatus, shown in Fig. 1, has been used with condenser energies up to 42 kJ with applied voltages up to 24 kV. Maximum neutron production of $\sim 3.5 \times 10^{10}$ neutrons per pulse has been observed; this corresponds to a production rate



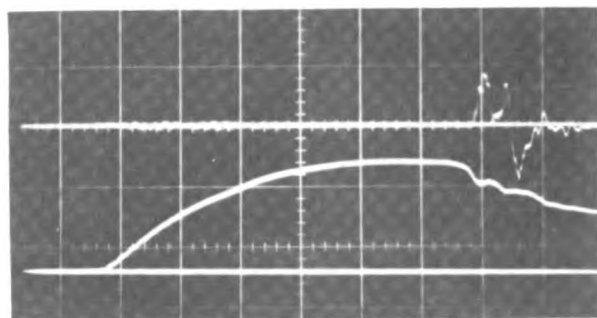
1



3



2



4

20 kV $P_{D2} = 5.9 \text{ TORR}$

$0.36 \mu\text{sec/cm}$

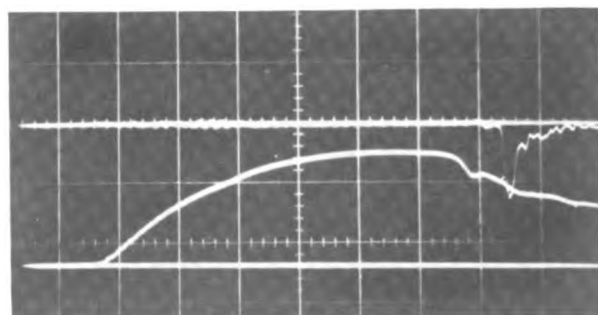
1.- $\frac{di}{dt}$, I, N~12

2.- V, I, N~9.6

3.- X-RAY (100-PIN-250) N~10.3

4.- X-RAY (PIN) & NEUTRON (PM), I, N~8.0

5.- NEUTRON (PM) ($\frac{1}{32} \text{ Pb}$) N~9.2



5

Fig. 2. Typical wave forms of current, (lower trace) with upper trace corresponding to $\frac{di}{dt}$ (1), Voltage (2), X ray (3), X ray and Neutron (4) and Neutrons (5) Sweep speed $0.36 \mu\text{sec/cm}$.

of $\sim 3.5 \times 10^{17}$ neutrons/sec of ≈ 2.45 MeV energy. Evidence to date shows that the neutron production increases with the value of stored condenser energy in an almost linear fashion.

The neutron production rate on a shot-to-shot basis is, under good conditions, within a factor of 2. Since the neutron rate depends strongly on the ion temperature, small fractions of partially ionized high Z contaminants can easily depress T_i and result in a drastic lowering of the neutron rate. A systematic study should be done to determine, for a given energy input, the best choice of electrode material and electrode shape to increase the neutron rate.

The result of using a deuterium-tritium (D-T) gas mixture instead of pure deuterium (D) showed an increase of $\sim 80 - 100$ in neutron production. The apparatus was essentially the same as in Fig. (1). The condenser bank, operated at 16 kV instead of 20 kV, gave a neutron yield of 14 MeV neutrons of $\approx 3 \times 10^{11}$ neutrons/pulse in a pulse time of $\sim 0.22 \mu\text{sec}$. The corresponding number for deuterium was $\sim 3.5 \times 10^9$ neutrons per pulse.

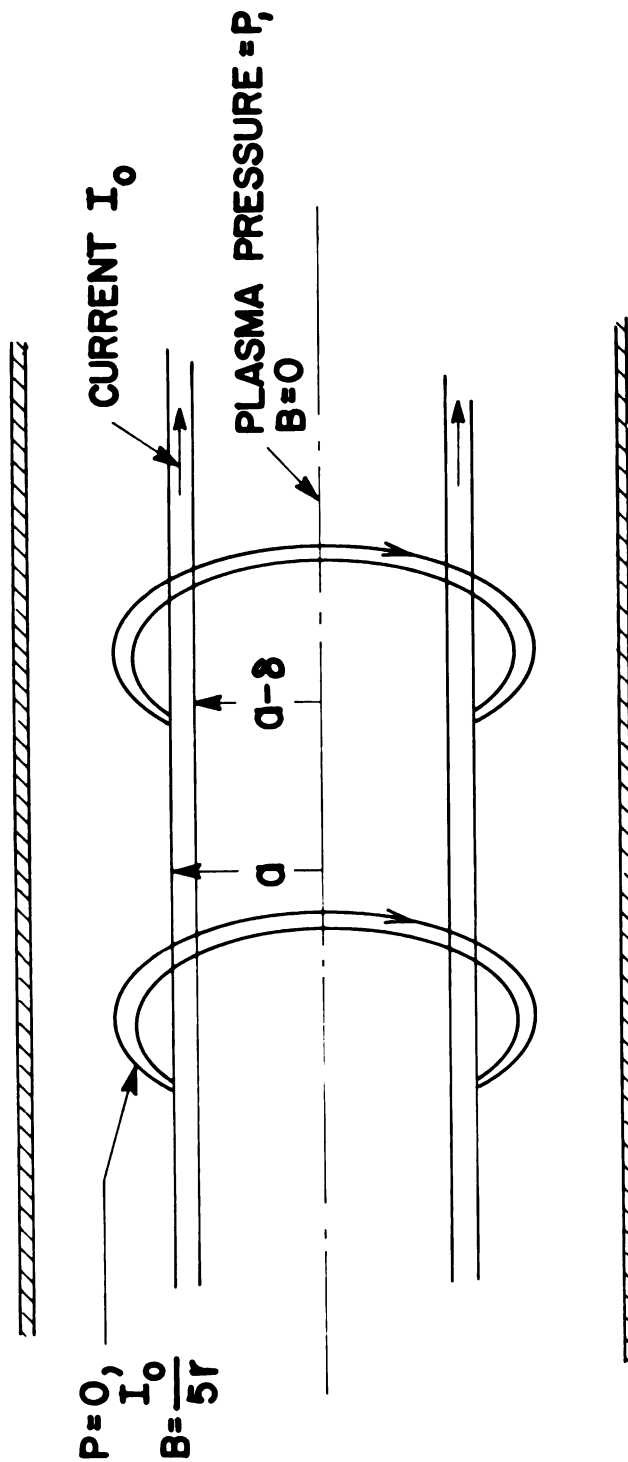
5. SCALING OF PLASMA FOCUS

As mentioned previously, the z-pinch effect seems to be the most probable mechanism for producing the hot dense plasma. The energy for collapse is provided by the magnetic energy that is inductively stored behind the current sheath at peak current time just before the sheath snaps off the end. The current sheath before collapse conforms almost to an ideal initial pinch situation, i.e., (1) the sheath carries a constant current, (2) the sheath is massive, i.e., the current sheath has "snowplowed" a major fraction of the gas it has encountered by the time it reaches the end, and (3) the electrical conductivity of the current sheath is high, insuring that the current flows in a relatively thin layer.

To illustrate the principle of magnetic confinement by the pinch effect first theoretically shown by Bennett⁶, let us adopt an oversimplified picture of the pinch and show the relation for the kinetic pressure balance between plasma and the pinch magnetic field.⁷ We assume, as shown in Fig. (3), that the current is confined to a thin cylindrical shell of outer radius a and thickness δ . Inside the shell, the plasma is of uniform particle density n_i and n_e and temperature T , and thus exerts a kinetic pressure $P = (n_e + n_i) kT$. Outside the shell, the plasma density and thus the kinetic pressure is equal to zero. The magnetic field within the radius $a - \delta$ is zero. The total inward force per unit area exerted by the shell is the integral of $j_z \times B_\theta$ through the shell, hence

$$\int_0^\delta j \times B \, dx = \frac{I_o^2}{2\pi a^2} \quad (1)$$

This force must just balance the kinetic pressure of the plasma $P = 2nkT$, i.e., $\frac{I_o^2}{2\pi a^2} = P = 2nkT$ where n is taken equal to $n_e = n_i$



SCHEMATIC REPRESENTATION OF
PINCH EFFECT IN A PLASMA

FIG. 3

or

$$I_0^2 = 2NkT \quad (2)$$

where N is the total number of particles per centimeter of length of pinch in a column of radius a. This relation is useful in establishing the current necessary for pinch confinement. A pressure balance between the magnetic field and the particle pressure can be written, using the relation $B = \frac{I}{5r}$ as

$$\frac{B^2}{8\pi} = 2nkT \quad (3)$$

where n and T is the final density, temperature and radius respectively.

If we assume that the plasma focus is an example of temporary confinement by a magnetic field, then we see immediately from Eq. (2) that the scaling of the total current results in larger NkT. To illustrate the strength of the magnetic field required for confinement, take the example of final density $n = 10^{19}/\text{cc}$, temperature $T = .5 \text{ keV}$ and final radius $r = .05 \text{ cm}$. From $B^2/8\pi = 2nkT$, one finds that $B = 2 \times 10^6$ gauss. For a plasma radius of 0.05 cm, the required current I is 0.5×10^6 amperes. So, in principle at least, the simple scaling of the condenser current should increase the final plasma pressure nkT.

To show how this is related to neutron production, one again assumes that the plasma focus is a confined plasma of high density such that interparticle collisions are effective in establishing a Maxwellian distribution. The total reaction rate for like ions, i.e., the D-D reaction, is $1/2 n^2 \langle \sigma v \rangle_{av}$ reactions per cm^3/sec where $\langle \sigma v \rangle_{av}$ is shown in Fig. 4 for both the D-D and D-T reaction to exhibit a strong temperature dependence. The reaction rate of D-D depends on n^2 and (in the temperature range of 2-10 keV) can be shown to a good approximation to depend on T_i^4 . So high ion temperatures as well as high densities are very important for the establishment of a hot dense plasma and large neutron production.

6. FUTURE POSSIBILITIES

There is apparently a limit to the amount of energy that can be used in a particular electrode system. The higher the energy or current, the greater is the effect of contamination as well as the physical destruction of materials due to larger magnetic pressures. Negating the above two effects, the pinch force on the plasma can in principle be made as large as desired because of its self compressing nature, i.e., one does not need to support magnetic fields of 10^6 gauss at physical walls.

Experiments are being designed in the 100 kJ range with currents up to $3-4 \times 10^6$ amperes. At the same time, the apparatus will be scaled in physical size i.e., larger electrode diameters will be used to reduce the current densities and magnetic fields in the accelerator section while at the same time taking advantage of the large current in the final radial collapse. An attempt will be made, using a small axial

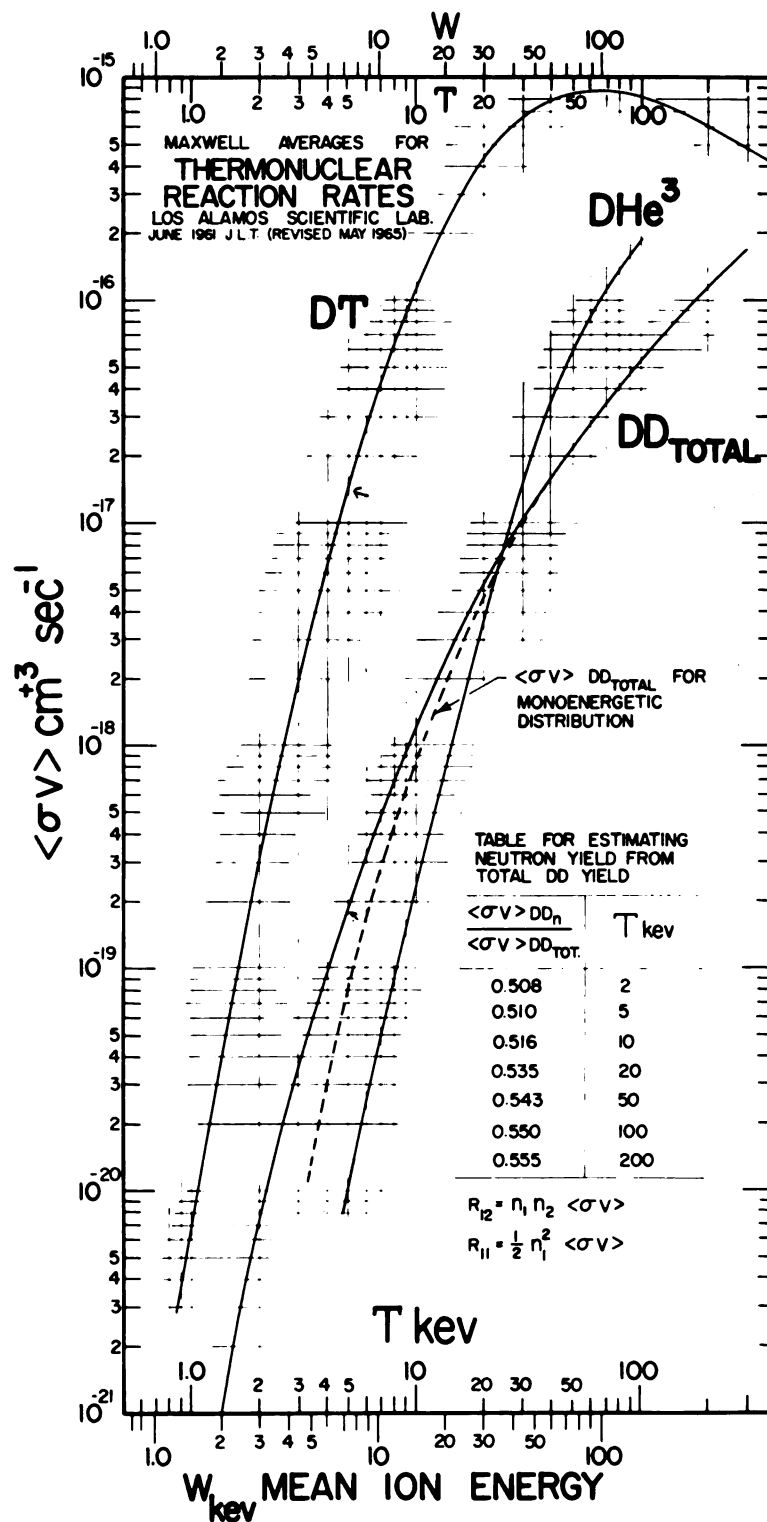


FIG. 4

magnetic field ahead of the center electrode to increase the stability of the plasma and hence to increase the plasma lifetime. Stability of such an axial current, in the ordinary z-pinch configuration has been a major problem in the controlled fusion project and it is not known at present whether the dense plasma focus exhibits a lifetime based on an instability or an electrode contamination effect. It appears that the plasma focus can be positioned off the geometrical axis by curving the accelerator to form a toroidal section. This scheme may offer the possibility of utilizing a multiple array of accelerators so that the plasma foci of several accelerators may be made overlap each other to produce a super dense focus in which the pinch current $I = \alpha I_0$ where α is the number of accelerators and I_0 is the current from each acceleration. According to Eq. (2), the final NkT would increase by α^2 .

On the basis of a simple energy extrapolation from the present energy source to ~ 250 kJ, one would expect neutron yields from D-D to reach 1.5×10^{11} neutrons per pulse. If the pulse width remains at ~ 0.15 μ sec, then rates of 10^{18} neutrons/sec can be expected, and correspondingly higher for the D-T reaction.

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DISCUSSION
OF
PAPER IV.D. (Mather)

Chairman: E. Bretscher

Secretary: B. Diven

KEEPIN: Your most interesting plasma device invites comparison to the somewhat similar Schmidt neutron flash tube -- similar at least in that it uses accelerated deuterons or tritons for the mechanism of production of neutrons. The thermonuclear plasma device is at least an order of magnitude greater in neutron yield per pulse. It has at least an order of magnitude shorter pulse width and, from what you have said about cost, it looks very economical. I can not help but compare it with the Schmidt neutron flash tube in cost per neutron. I suspect your source beats the flash tube on all counts -- yield per pulse, pulse width, and perhaps even on cost. That latter point is what I would like to ask Beckurts about.

BECKURTS: Well, I think you are right. With the flash tubes the best we have made so far is 10^9 neutrons per burst in about 1 μ sec. I think this can be improved by only a factor of 10 with such short bursts, and even this may cause considerable difficulties. Of course, the flash tube could operate at much higher repetition rates. Up to 200/sec or so have been realized so far, but I really believe that this is a different order of magnitude. Of course the physics is quite different because this is a true thermonuclear device, whereas the flash tube is essentially an accelerator. A flash tube giving 10^{11} neutrons per second average production rate might cost about \$50,000.

KEEPIN: Now for comparison, I would like to inquire of Mather further about the actual capital cost of the plasma device. (I believe you mentioned a figure of \$3,000 -- was that just for the power supply and condensers, or what?) From the standpoint of a university or some small research organization, for roughly \$50,000 you can install and start using a flash tube. What would be the comparable capital cost of a plasma accelerator, and would it be available to, say, a university?

MATHER: Well, I did not really press the comparison with all accelerators. Because you do not make these neutrons by a direct acceleration, the cost is a few thousand dollars versus much more for an accelerator. (Operating costs are very low.) The capital investment for condensers alone is about 10¢ a joule, so installed I would give a figure of 20-25¢. I think this experiment could be set up again in a small laboratory for maybe \$10,000 including pumps and vacuum chambers; it is a very simple apparatus, and this device is available to anyone. The other thing I wanted to stress is the fact

that this source is monoenergetic, that is, you are dealing with 2.45 MeV neutrons from the (d,d) and 14 MeV neutrons from the (d,t). It does not have a fission spectrum or a continuous spectrum -- it has these two discrete energies.

MILTON: I would like to point out in line with Taschek's remarks this morning that we are talking about intense neutron sources here and whether you try to produce an average continuous flux of 10^{16} n/cm²/sec by a reactor or by almost any other kind of device, you are going to have to produce 10^{19} n/sec. Now, according to some quick arithmetic I did here, that means scaling up the example in Fig. 1 by a factor of 3×10^8 in order to go from 3.5×10^{10} n to 10^{19} n/sec. Therefore, this device is not really to be considered as a source for a continuous high flux device.

MATHER: (Note Added In Proof: A high flux device can have either a high average rate of neutron production or a high instantaneous rate. The thermonuclear plasma with the DT source has demonstrated 10^{19} n/sec during the burst and can be extrapolated easily to 10^{20} n/sec instantaneous production rate. The existing high flux accelerators such as the Nevis synchrocyclotron work at 10^{19} n/sec instantaneous. The thermonuclear plasma in its present form is a device with a high instantaneous neutron production rate and a low average rate. Today's simple plasma machine cannot compete with the proposed ING machine but it can equal the instantaneous rate of any existing accelerator.)

BOLLINGER: I just wanted to make sure I understood the question of cost also. As I understand it, the statement was that one got 10^6 n/burst to a quarter dollar capital investment. Was that a correct statement?

MATHER: Let us say the machine in-house cost \$10,000 excluding shielding, just to take a round figure, to produce the 3.5×10^{10} (d,d) n/burst. Well, it is just 3.5×10^{10} divided by \$10,000, so many n/dollar/burst. So I think it is quite cheap. At one burst/sec, the (d,d) source gives 3.5×10^{10} n/sec and the (d,t) source gives 3×10^{12} n/sec at a cost of \$10,000 or 3.5×10^6 and 3×10^8 n/sec/dollar, respectively. Existing accelerators used as slow neutron spectrometers produce neutrons at an average rate of 10^{13} /sec at a cost greater than 10^6 dollars or fewer than 10^7 n/sec/dollar. Of much more importance in considering the use of this simple device is its great peak intensity that can be easily scaled up to 10^{20} n/sec which is greater than any existing machine listed by Havens in his chart. The immediate application of this plasma device will not be in producing large average neutron output, but rather in applications needing the intense pulse. (Editor's Note: The thermonuclear machine does not give only one burst as might be inferred from the above cost accounting. The first burst would cost this much, consequent bursts require only charging currents.)

SESSION V

ACCELERATORS AS INTENSE SOURCES OF THERMALIZED NEUTRONS

SPALLATION-TYPE THERMAL NEUTRON SOURCES

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Three laboratories have an active interest in spallation-type intense thermal neutron sources. At Oak Ridge a program of development of the Separated Orbit Cyclotron¹, has as one objective acceleration of a high current proton beam for neutron production^{2,3}. Preliminary studies have been made by a group under Basil Rose at Harwell⁴ to consider the feasibility of a high flux neutron source using a high current proton accelerator and there are close ties between that group and the high current SOC development program under F. M. Russell at the Rutherford Laboratory⁵. The Intense Neutron-Generator project at Chalk River has reached the stage where a proposal has been presented to proceed with development with 1974 as the target date for operation. This report will be concerned mostly with ING where conceptual design of the thermal neutron target is well advanced. A detailed description of the ING project will be found in the proposal report⁶. The accelerator included in the reference design has recently been switched from the SOC to the linac for reasons to be discussed later.

A presentation of the thermal neutron capabilities of spallation-type sources and comparison with the capabilities of other devices for thermal neutrons would be incomplete if the wide scope of the spallation facility in other applications was ignored. The breadth and versatility of ING as presently conceived is shown in Fig. 1 which also portrays the evolution of the project from initial needs for higher thermal flux, on the left to future programs outlined on the right. The original motivation behind this study was to provide a high continuous flux of thermal neutrons of the order of $10^{16} \text{ cm}^{-2} \text{ sec}^{-1}$. This is desired by the CRNL solid state group who would like this average flux for inelastic neutron scattering experiments. They have yet to be convinced that pulse techniques will be as powerful or convenient for their experiments if they have to accept a reduction in average flux. Other experimenters also had good reasons for continuous flux, for example, for multiparameter neutron capture γ -ray experiments requiring very pure thermal beams and to produce an intense variable energy line source of γ -rays for γ -ray resonance excitation experiments. Others are interested in fundamental experiments such as a better measure of the neutron lifetime and various experiments using a neutron gas target. In addition there was a motivation to produce radioactive isotopes as suggested in the figure. All of these applications require a continuous high flux source and would not be well satisfied by a pulsed source. But as will be discussed later, we can conceive of a way to pulse the target without loss

of average flux if this ultimately becomes desirable. The reasoning behind the choice of the spallation reaction to achieve a CW flux of $10^{16} \text{ cm}^{-2} \text{ sec}^{-1}$ have been presented in detail elsewhere^{7,8,9} and need not be belaboured here. It is safe to say that, if the goal of this project were solely to produce a high flux, it probably would be easier and cheaper to build a high flux reactor. The spallation route was chosen at CRNL because the neutrons are produced more efficiently than in a reactor - as measured in terms of the neutrons emitted per unit of heat generated in the source - and therefore the method is intrinsically more promising than a reactor for extension to still higher fluxes at a future time. As shown in Fig. 1, in addition to a high flux of thermal neutrons for the various applications in research and isotope production, ING provides a means of making neutron-poor isotopes and of generating beams of energetic nucleons, π and μ mesons and neutrinos. The latter beams can be produced in both pulsed and CW form and over a wide range of energies. Initial experiments with a zero-energy advanced power technology target may open the door for development of power/breeder targets and electronuclear power systems. There are further practical advantages in stimulation of new research and development as indicated in the figure. This presentation is intended to emphasize the fact that in comparing this type of device with others giving a high flux of thermal neutrons the question of scope and versatility in other applications must also be considered. A further application not stressed in Fig. 1 but favoured strongly in the Oak Ridge planning is the use of intermediate energy beams at 50 MeV for nuclear physics research.

Let us then next consider briefly the planned layouts for the ORNL and Chalk River facilities. The Harwell study has not as yet produced a layout plan for an intense neutron source. The Oak Ridge facility³ is shown in Fig. 2. This layout gives maximum opportunity for using the first stage, 50 MeV, SOC for nuclear physics research while construction proceeds on the later stages. The phase-one facility is enclosed by the heavy black line. They plan to use the 50 MeV stage for acceleration of either H^- or H^+ ions. The H^- beam passes through a high-resolution analyser and into a tandem to give a proton beam with high resolution and fine energy control. The circular room at the center contains a high-resolution reaction products analyser. The facility can be expanded in three phases ending with the 1000 MeV SOC stage. This could supply a beam of the order of 75 mA to the Super Flux Thermal Neutron Facility shown on the left.

An overall plan of the Chalk River project is shown in Fig. 3. The linac, some 5000 ft long, is shown to the right. From the beginning, the SOC and the linac have been the two most likely contenders as the accelerator required to produce the 65 mA needed for the high-flux facility. Until very recently the ING reference design included the SOC and most of the analysis and design effort of the Chalk River accelerator team were devoted to it. But there was also a working party keeping informed of linac developments at Los Alamos and elsewhere. Very recently, indeed since the

final manuscript of the ING proposal was written, the decision was made to switch to the linac as the reference design accelerator, not because any flaw had suddenly been discovered in the SOC, but because the overall view looked brighter for the linac. Briefly the decision boiled down to two main considerations: 1) assurance that the device can be lined up magnetically within tolerances required to avoid excessive beam spill in the accelerator itself and 2) the availability of an RF system, including power amplifiers and accelerating structures, that would give the highest possible AC power to beam power efficiency. The difficulty in achieving the necessary confidence concerning the ion optics and beam spill in the SOC prior to actual full scale construction on the one hand, and the development of highly efficient wave guide structures for the linac at Los Alamos and an increasing confidence in Amplitrons as highly efficient dependable RF power amplifiers on the other, tipped the scale in favour of the linac.

Figure 3 shows, on the left, the experimental areas for two phases of construction. The first phase shown by the hatched areas includes a small meson experimental area and at the end of the tunnel, the thermal-neutron facility. The second phase would include a larger meson and fast neutron area and a split beam channel leading to a resonance neutron time of flight facility and the advanced power technology area. These areas and the flight paths are shown by broken lines. Other buildings are for offices and laboratories, services, AC power switchyard and so forth.

The experimental areas for the full machine are shown in more detail in Fig. 4. This layout does not exactly agree with Fig. 3 but the differences are of no consequence. The philosophy behind this layout was to keep all parts as spread out as possible consistent with keeping beam transport costs as low as possible. It is desirable to keep things spread out to reduce background problems and allow room for additional shielding if necessary, or for making modifications or additions for new applications which cannot be foreseen. The full 65 mA beam enters the High Activity Tunnel at the bottom of the figure. It passes through a beam splitter which diverts ~ 1 mA 90° left into beam tunnel No. 2 and up to 100 μ A 45° right into tunnel No. 3. The remainder of the beam (roughly still 65 mA) passes on to the Thermal Neutron Facility. In the High Activity Tunnel the main beam passes through two thin targets for producing mesons. The High Activity Tunnel will require remote handling equipment. This tunnel and tunnel No. 2 are shielded by 35 ft of sand held by cement walls. The Thermal Neutron Facility Tunnel and tunnel No. 3 are covered by sand mounds also 35 ft thick.

Beam No. 2 passes through various other meson targets, shown dotted in Fig. 4, and ultimately strikes a liquid D_2 target for fast neutron production. The spent proton beam is deflected to an underground beam dump. Beam No. 3 passes through a second splitter which deflects 0.1 μ A to the advanced power technology target area. Most of beam No. 3 is delivered to the Resonance Neutron Time of Flight Facility.

An elevation view of the Thermal Neutron Facility is shown in Fig. 5. The

proton beam is delivered to the top of the shield whence it passes through a bending magnet and is directed downward onto a target of molten Pb-Bi eutectic. It is unnecessary to have a window in the beam tube at this point because the vapour pressure of Pb-Bi is less than the 10^{-6} torr vacuum required for the accelerator and beam transport. The Pb-Bi is circulated through a heat exchanger below floor level. The Pb-Bi target is surrounded by a moderator vessel penetrated by beam tubes, very much like a heavy-water reactor. Around the moderator is a 15 ft-thick shield of iron and heavy concrete. The neutron beam tubes are enclosed in pipes as a personnel safety precaution. The pipes would normally terminate at experimental apparatus followed by a beam stop which is a massive object and may, as shown here, be located outside the building.

The target arrangement is shown in more detail in Fig. 6. The Pb-Bi eutectic enters at the top where vanes give it a vortex motion which forces it to cling to the walls of the target tube. It forms a surface at the center of the tank where the beam strikes it. The target is surrounded by a Be multiplier which increases the neutron output by the (n,2n) reaction.

Figure 7 shows a plan view of the beam tubes and irradiation facilities. There are seven beam tubes. Five 10 cm diameter tubes and one 20 cm diameter through-tube give access to the flux of $10^{16} \text{ cm}^{-2} \text{ sec}^{-1}$ at 12 cm from the Be multiplier surface. In the same circle are two pneumatic rabbit facilities for isotope irradiation. Many of the horizontal tubes have vertical access pipes. These permit installation of targets on other objects in the high flux region without obstructing the beam tube proper, are otherwise available independently as irradiation facilities, and supply a rigid support for the end of the beam tube. Any tube, vertical or horizontal, not in use will be flooded by circulating D_2O to improve the moderator efficiency. At a radius of 76 cm there are two 15 cm-dia vertical irradiation thimbles and one 10 cm-dia tangential tube with vertical access. At a radius of 96 cm there are three 15 cm-dia vertical thimbles. At about 110 cm radius there is a circle of 45 vertical hydraulic capsule facilities for isotope irradiation. Near the tank wall is a double annulus containing cobalt in the form of small 'fluidized' pellets. On the floor of the tank are six hydraulic capsule facilities as shown in the diagram on the right of Fig. 7.

Table I gives a few pertinent dimensions and parameters of the Thermal Neutron Facility. The proton beam current and energy have been fixed by detailed investigations of the neutron yields from spallation in heavy elements and the neutron moderation in the target-moderator assembly. The neutron yields and heat output were studied by detailed Monte Carlo calculations by Milton and Fraser at CRNL¹⁰. The neutron yields and flux distribution were studied experimentally by a joint team from Chalk River and Oak Ridge¹¹ using the BNL Cosmotron. Good agreement between theory and experiment was obtained. The neutron yields obtained by foil activation are believed to be accurate to $\pm 5\%$. The agreement between calculated and measured yields was somewhat poorer but

the reasons for this discrepancy are believed understood and it would seem safe to assume that neutron yields for any similar geometry of lead target can now be estimated with an accuracy of 5-10%. The fast, epithermal, and thermal fluxes for a lead target and D₂O moderator have been obtained by Kushneriuk and others¹² at CRNL by solving the multigroup diffusion equations in cylindrical geometry. These calculations have been repeated and verified at Harwell⁴. The calculations did not take into account the multiplication and moderation by the beryllium multiplier or the flux depression effects of the target tube and beam tubes. The latter question has been studied experimentally with a simulated target and D₂O tank assembly at Chalk River¹³. The performance of the multiplier has yet to be checked by a mockup experiment but, from rough calculations, it is concluded that the volume of D₂O in which the 10¹⁶ flux is obtained will be substantially increased by the presence of the beryllium.

TABLE I

ING THERMAL NEUTRON FACILITY

Beam Tube Maximum Flux (perturbed)	10 ¹⁶ n cm ⁻² sec ⁻¹
Source Strength	10 ¹⁹ n sec ⁻¹
Target Material	Pb-Bi eutectic, Zr tube, Nb liner
Target Dimensions	20 cm dia x 60 cm effective length
Moderator	D ₂ O
Moderator Tank	120 cm radius
Shield	Iron, heavy concrete
Shield Thickness (midplane)	450 cm (iron)
Proton Beam	65 mA, 1 GeV, CW
Beam Power	65 MW
Target Power	38 MW
Moderator Power	16 MW
Shield Power	11 MW

Table II gives the fluxes and approximate Wescott r-values for various beam tubes and irradiation facilities. The calculated flux distribution at the surface of the target in the presence of the moderator peaks at 0.03 eV and has a roughly 1/E tail from 0.5 eV to 1 MeV.¹² Above 1 MeV the spectrum appears as shown in Fig. 8. The peak at 3 MeV is caused by a combination of the original evaporation spectrum which peaks near this energy and the effect of the 0¹⁶ scattering cross-section which has a dip at this energy. The uncollided cascade neutron spectrum is responsible for the shape above 20 MeV.

The intensities of various groups⁶ are shown in Fig. 9. Although these curves were calculated for somewhat different target dimensions, they will give a good representation of the spectrum as a function of position in the moderator. The effect of the multiplier is not included; it can be expected to improve the thermal to fast ratios over those presented here. At 12 cm from the target face (or from the multiplier face in the ING assembly Fig. 7) where the beam tubes are situated, the ratio of thermal neutrons to neutrons >1 MeV is $\sim 30:1$. For HFBR this ratio is $\sim 100:1$; about 30 times better than in a radial tube looking directly at the core¹⁴. From this we may conclude that the thermal-to-fast ratio in the tangential beam tubes of ING will be about ten times better than that in the radial tubes of a reactor. However, ING will be at a disadvantage in the flux of very high energy neutrons >20 MeV. Some of these will find their way out of the beam tubes. Those in the range >200 MeV are very penetrating and require massive shielding and beam-stops to reduce their intensity to biological tolerance levels.

Before leaving the thermal neutron facility I would like to discuss its potential for extension to fluxes higher than $10^{16} \text{ cm}^{-2} \text{ sec}^{-1}$. There are two possibilities: a) increasing the CW flux and b) pulsing the target.

Increase of source strength in CW operation is limited by i) the cost of beam power and ii) the necessity of removing the local heat generated in the target. I will ignore the economic question and, give only an order-of-magnitude estimate of the ultimate performance permitted by the physical limitations. We assume that heat transfer with the Pb-Bi eutectic limits useful power densities to $\leq 10 \text{ MW/l}$. Since almost all the heat is developed in the first 100 cm of the target, the average power density of the present design is roughly 1 MW/l and could therefore be increased by a factor of ten. We can also increase the diameter of the target with advantage until it approaches the migration length at about 100 cm after which increases in size will have little effect on the flux. Both together would increase the source strength (an incidentally the beam current) by a factor of 250. The flux gain would not be exactly proportional but we can conclude that the hypothetical physical limit for an ING-type source lies somewhere near $10^{18} \text{ n cm}^{-2} \text{ sec}^{-1}$. This is roughly an order of magnitude greater than that projected for a homogeneous reactor¹⁵.

TABLE II

THERMAL NEUTRON EXPERIMENTAL AND IRRADIATION FACILITIES

<u>Facility</u>	<u>Diam</u> <u>cm</u>	<u>Quantity</u>	Max Thermal Flux Unperturbed $\text{cm}^{-2} \text{sec}^{-1}$	Westcott r Approx.
Horizontal tangential "L-tube"	10	1	10^{16}	0.026
Horizontal tangential "L-tube"	10	1	5×10^{15}	0.003
Horizontal tangential stub	10	1	10^{16}	0.026
Horizontal radial stub	10	1	10^{16}	0.026
Horizontal tangential "T-tube"	20	1	10^{16}	0.026
Pneumatic rabbits	10	2	10^{16}	0.026
Vertical irradiation thimbles	15	2	$\sim 5 \times 10^{15}$	0.003
Vertical irradiation thimbles	15	3	$\sim 1 \times 10^{15}$	---
Hydraulic capsule facilities	4	45	$\leq 10^{15}$	---
Inner cobalt blanket 3/8" thick		1	$\leq 10^{15}$	---
Outer cobalt blanket 3/4" thick		1	$\leq 10^{15}$	---

Pulsing of the thermal source can be achieved by pulsing the proton beam. The most attractive possibility is to store the 65 mA CW beam from the accelerator in a storage ring which may be discharged at intervals into the target. Very preliminary considerations of the physical requirements of this device encourage the belief that 0.15 μsec -long bursts at 500 per second may be feasible. This would increase the instantaneous neutron source intensity in the thermal target by 1.3×10^4 . Because of the smearing effect during moderation, not all of this gain is available at thermal energies. However, it has been estimated⁷ that in a 5 μsec interval at the peak of the thermal pulse ($E < 0.13 \text{ eV}$) the flux would be about $10^{17} \text{ n cm}^{-2} \text{ sec}^{-1}$ or a gain of a factor of 10. The gain for epithermal neutrons ($0.13 < E < 1.46 \text{ eV}$) would be greater, perhaps a factor of 100. To make use of these pulses for time-of-flight one would need a mechanical shutter with a 5 μsec burst width synchronized with the pulse from the moderator.

The fast neutron facility; beam-2 Fig. 3 will receive from the splitter $\sim 0.7 \text{ n sec}$ long bursts of protons with an instantaneous current of 560 mA or 2×10^9 protons per burst. This burst width is based on the assumption that the linac will operate at 200 MHz in the Alvarez section and 800 MHz in the waveguide section. The burst length takes account of spread introduced by the long proton transport system. With a liquid D_2 target 40 cm thick, and a pulse rate of 2 MHz it will be possible to obtain an essentially monoenergetic 1000 MeV pulsed neutron beam with a flux of $6 \times 10^5 \text{ cm}^{-2} \text{ sec}^{-1}$.

at 180 ft or, with a 400 gm cm^{-2} Bi target, some 2.5 times that flux would be obtained for neutrons of all energies $>200 \text{ MeV}$.

The beam delivered to the resonance neutron facility will also consist of micropulses with a width of $\sim 0.7 \text{ n sec}$. Given a versatile beam splitter system these can be delivered to the target in groups (macropulses) with any desired pattern, i.e., the number of micropulses in a macropulse and the repetition rate of the macropulses can be varied as required. This feature makes it possible to optimize the pulse structure for the energy range under study. To illustrate the performance we have calculated the flux available at the end of the flight path in the resolution width equal to the Doppler width at energy E_n under optimum conditions. This is perhaps the most relevant figure of merit for a resonance neutron facility. The flux is plotted as the curve $F(\text{MP})$ in Fig. 10. The Doppler width corresponds to a target of mass number 100. The curve $\ell(\text{MP})$ shows the flight path required at each energy. This figure was drawn for the SOC microstructure (50 MHz, 2.4 n sec), but the curves are the same for the linac because the average current is the same and we select the macropulse width as before. The repetition rates required are less than 2000 per second at all energies. At 10 eV the macropulse would contain 126 micropulses; above 100 keV the number is reduced to one. Since fluxes as low as $10^{-4} \text{ cm}^{-2} \text{ sec}^{-1}$ are useful for transmission measurements it would appear this device will permit measurements with Doppler resolution up to 300 keV or higher. The curves are shown only for flight paths less than 1 km.

The same figure shows similar curves for the storage ring. Here it is assumed that the target resembles the thermal neutron target in Fig. 6 but the shape of the surrounding moderator is optimized for resonance neutron production. The repetition rate is 500 pulses per second, the pulse width 150 nsec. With this arrangement, 1 km flight paths are required at about 8 keV but the flux available at the end of the flight path is three or more orders of magnitude greater than for the micropulse facility.

From this discussion I think it is evident that the fields of applicability of spallation-neutron sources like ING is very large and that the intensities and pulse characteristics are favourable for a varied research program in neutron physics.

I wish to acknowledge the contributions to the ING program⁶ of the Chalk River scientists and engineers who are too numerous to mention by name.

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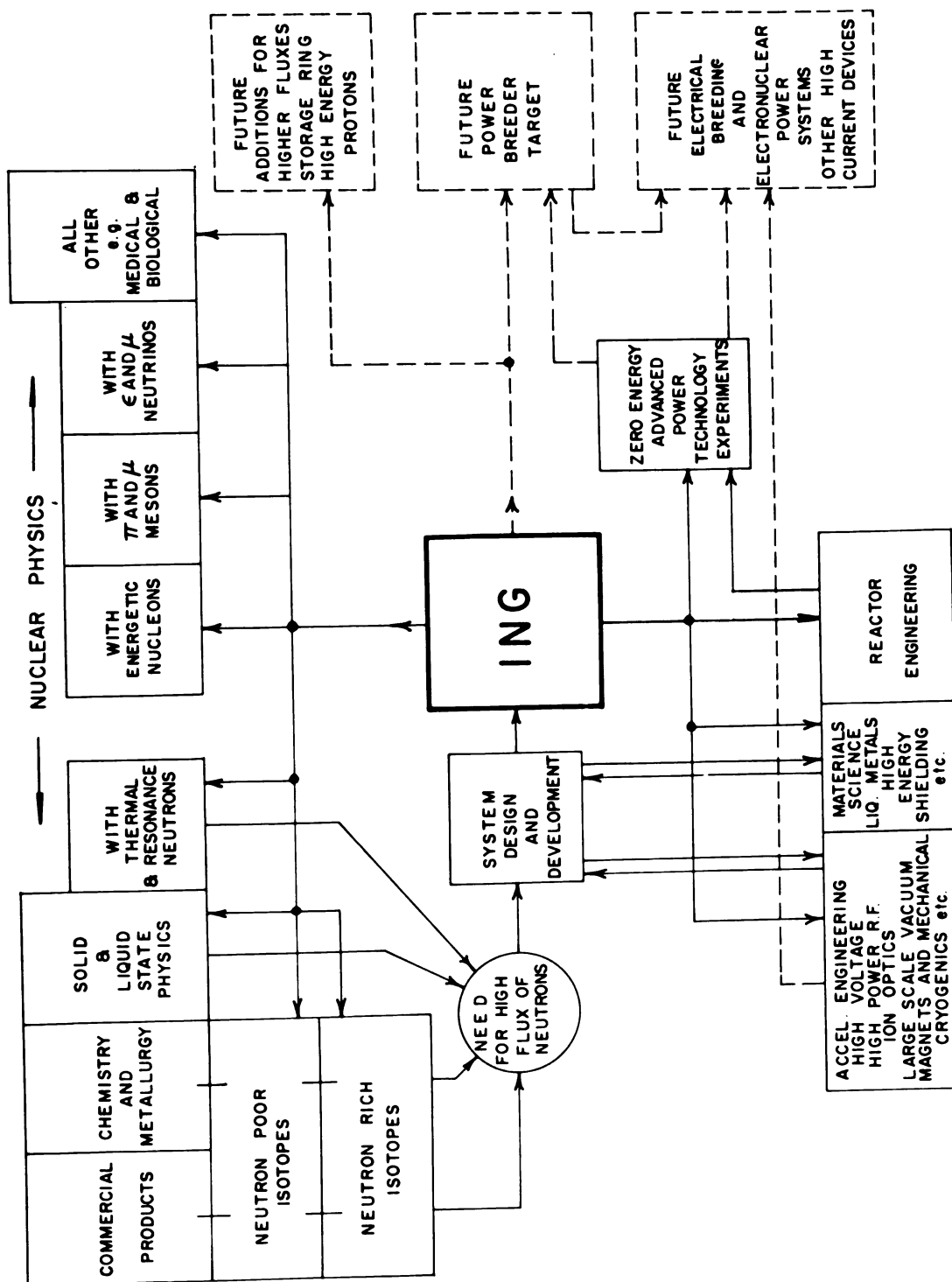


Fig. 1 Evolution and Application of ING System.

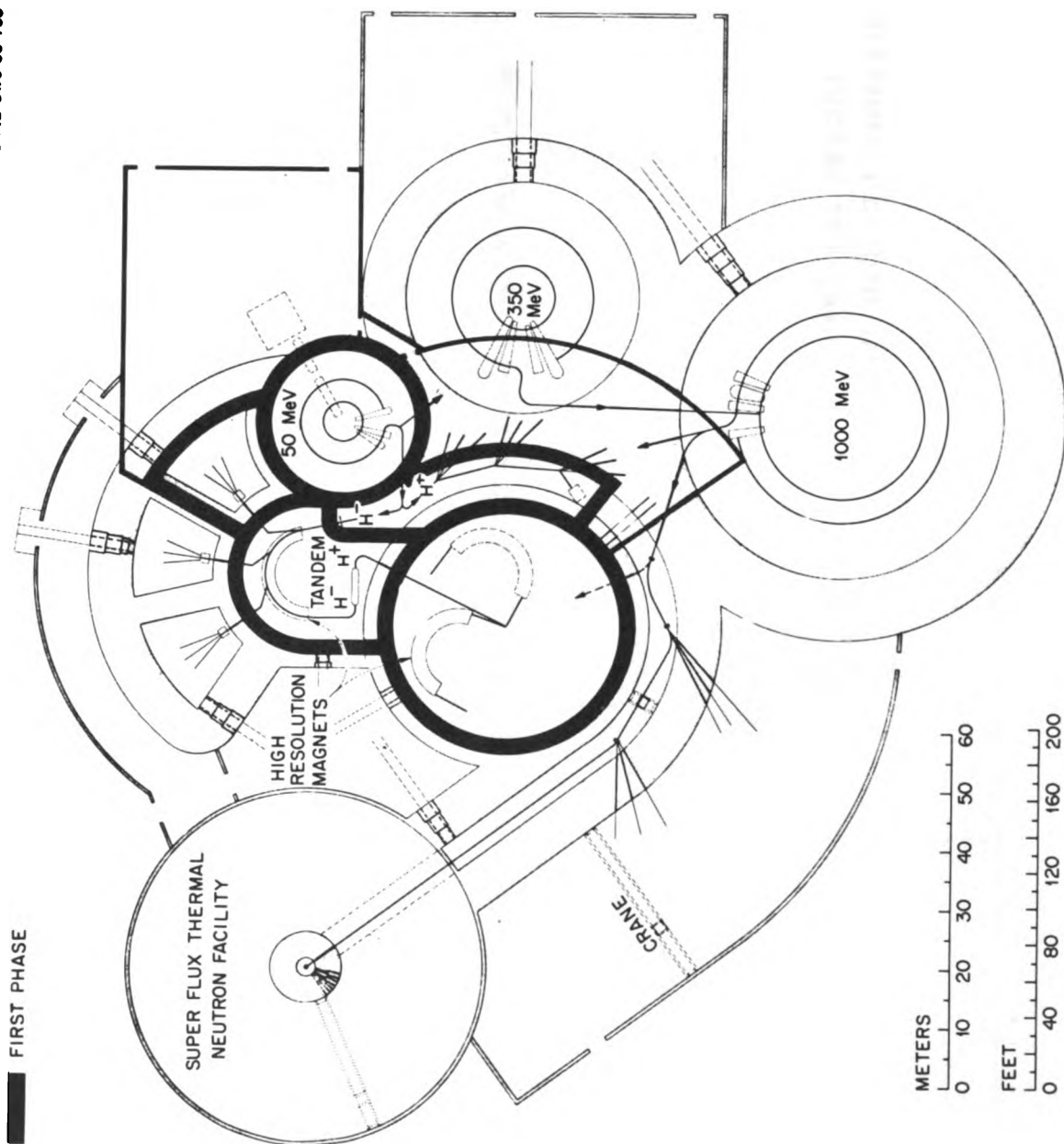


Fig. 2 ING Complex

INTENSE NEUTRON GENERATOR LINAC SITE LAYOUT

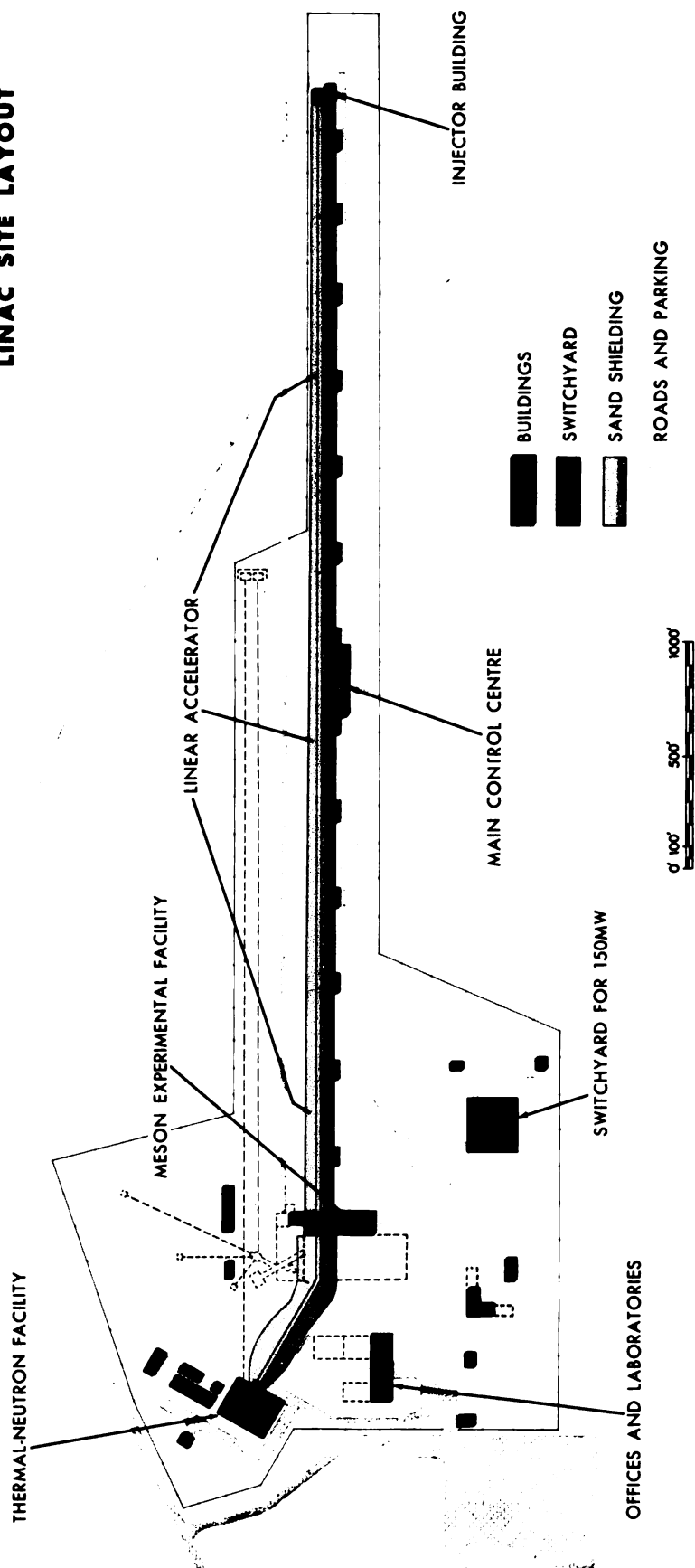
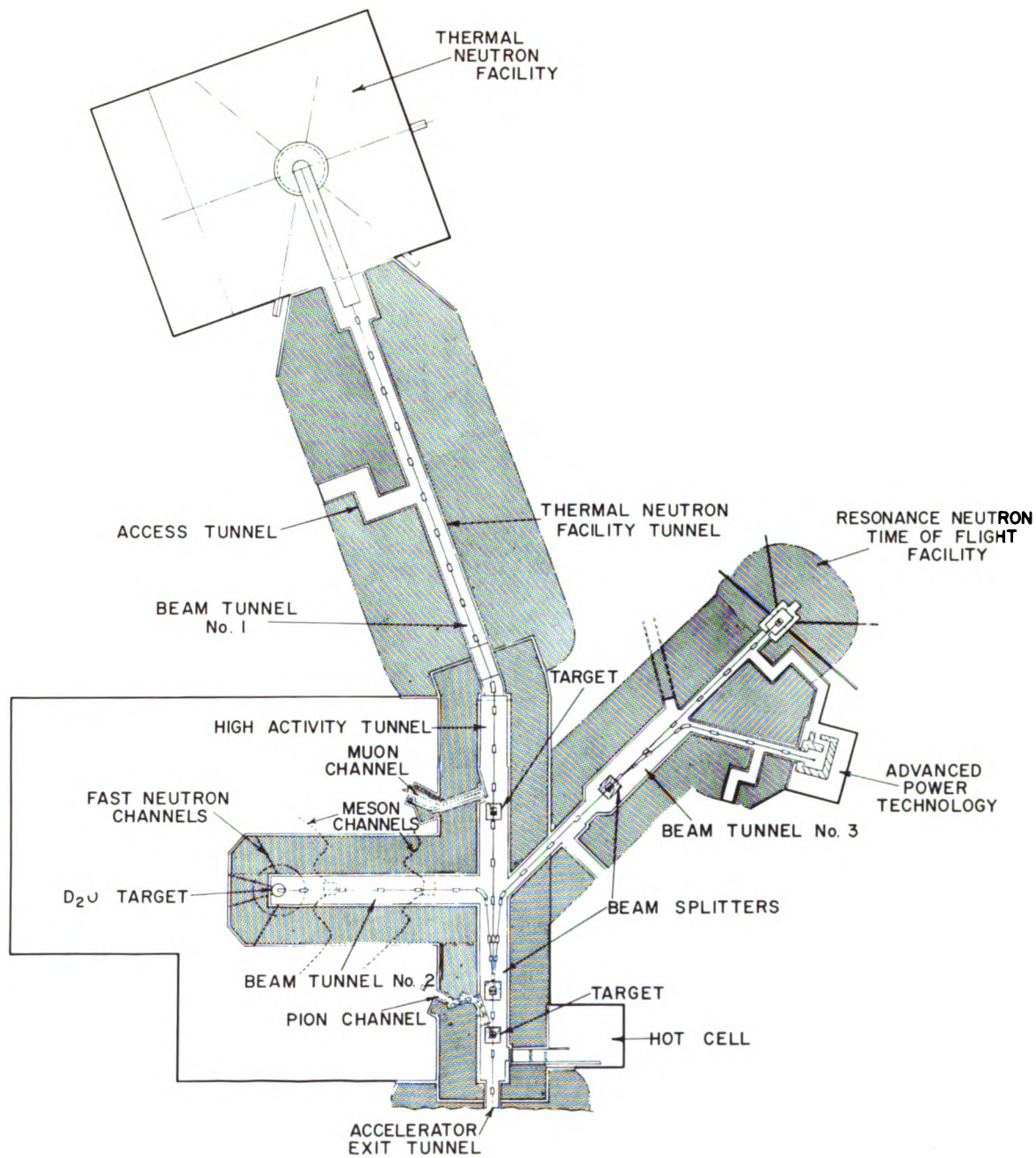
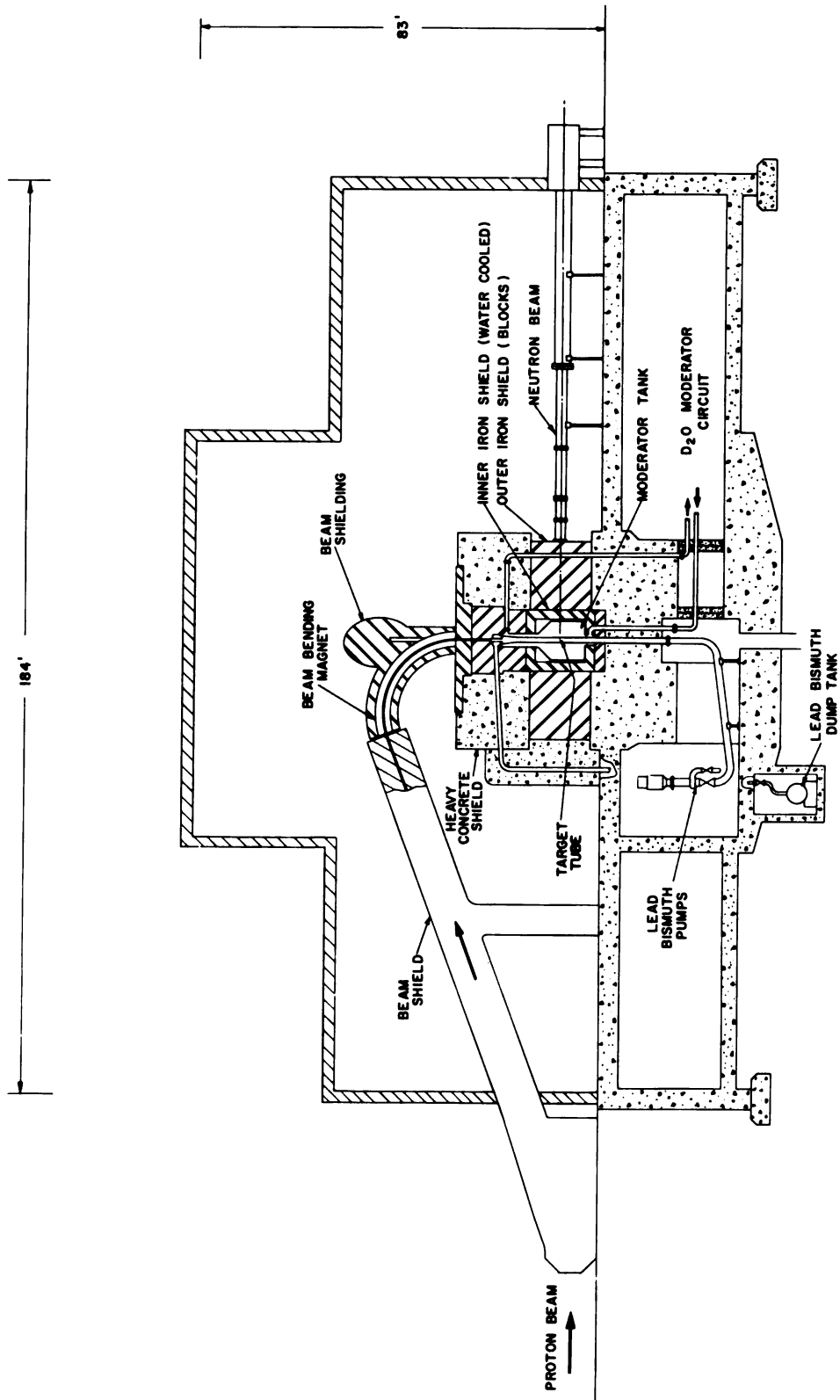


Fig. 3 ING Transport and Experimental Area Layout⁺



BEAM TRANSPORT SYSTEM

Fig. 4 ING Transport and Experimental Area Layout



CROSS SECTION THERMAL NEUTRON TARGET

Fig. 5 Thermal Neutron Facility Elevation

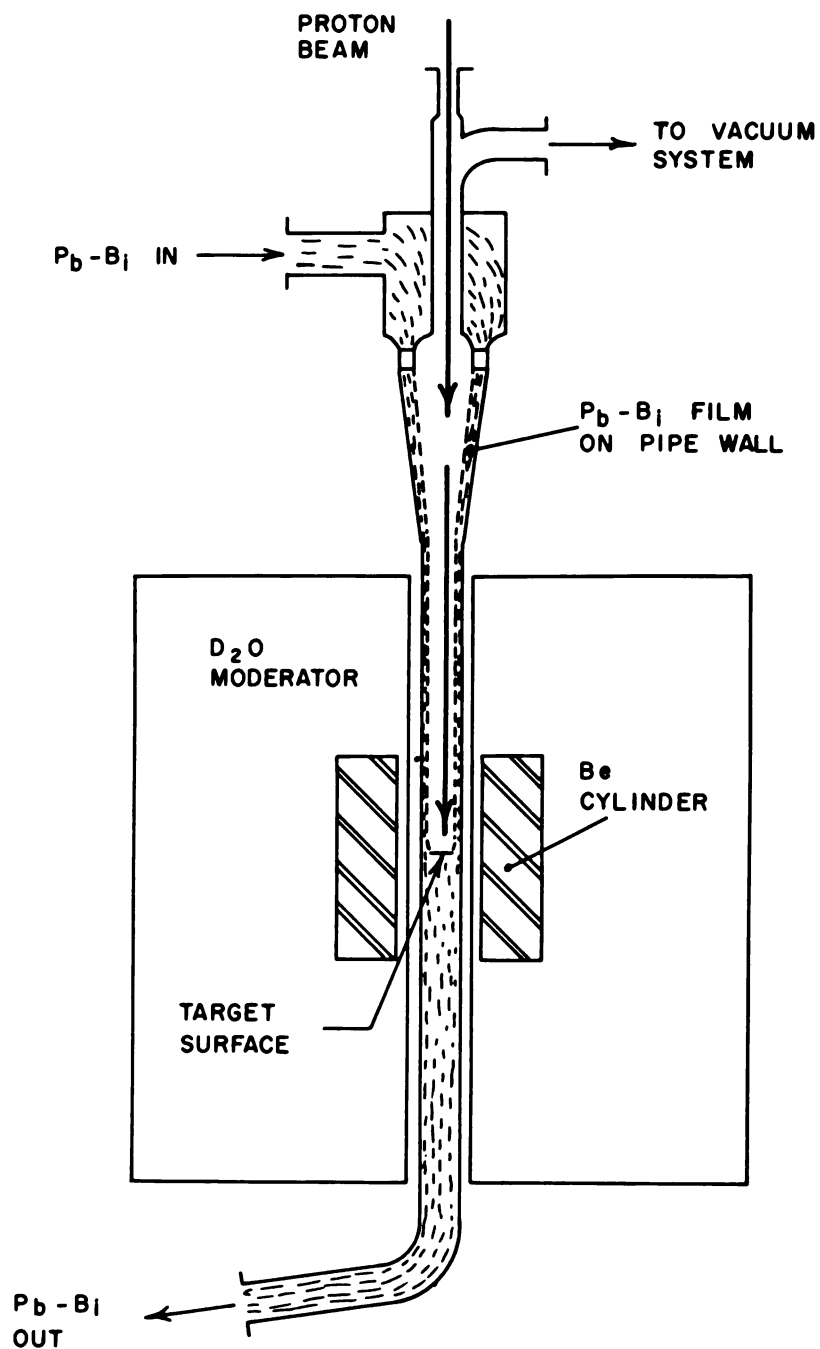


Fig. 6 Thermal Neutron Target Elevation

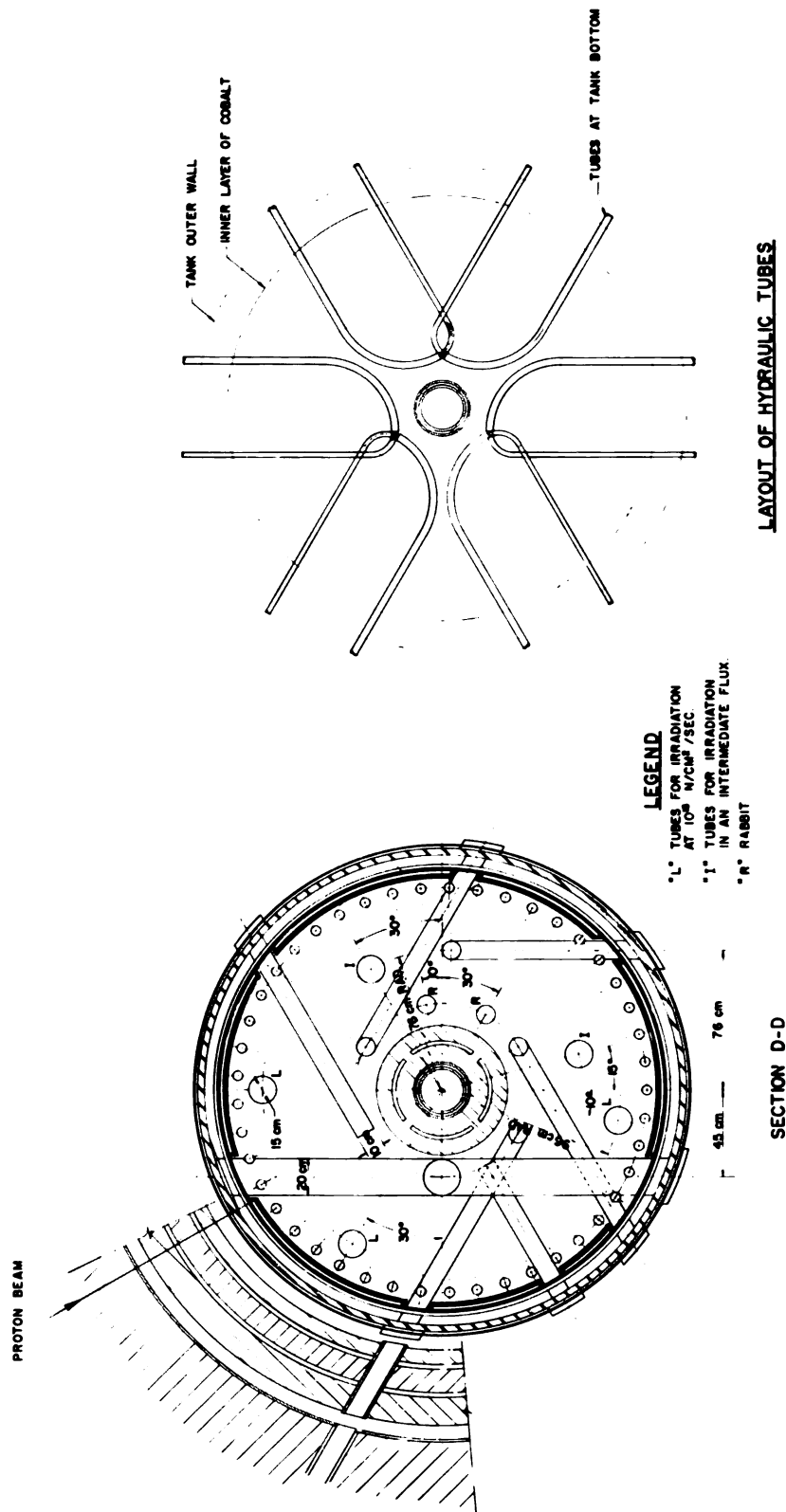


Fig. 7 Thermal Neutron Target Plan

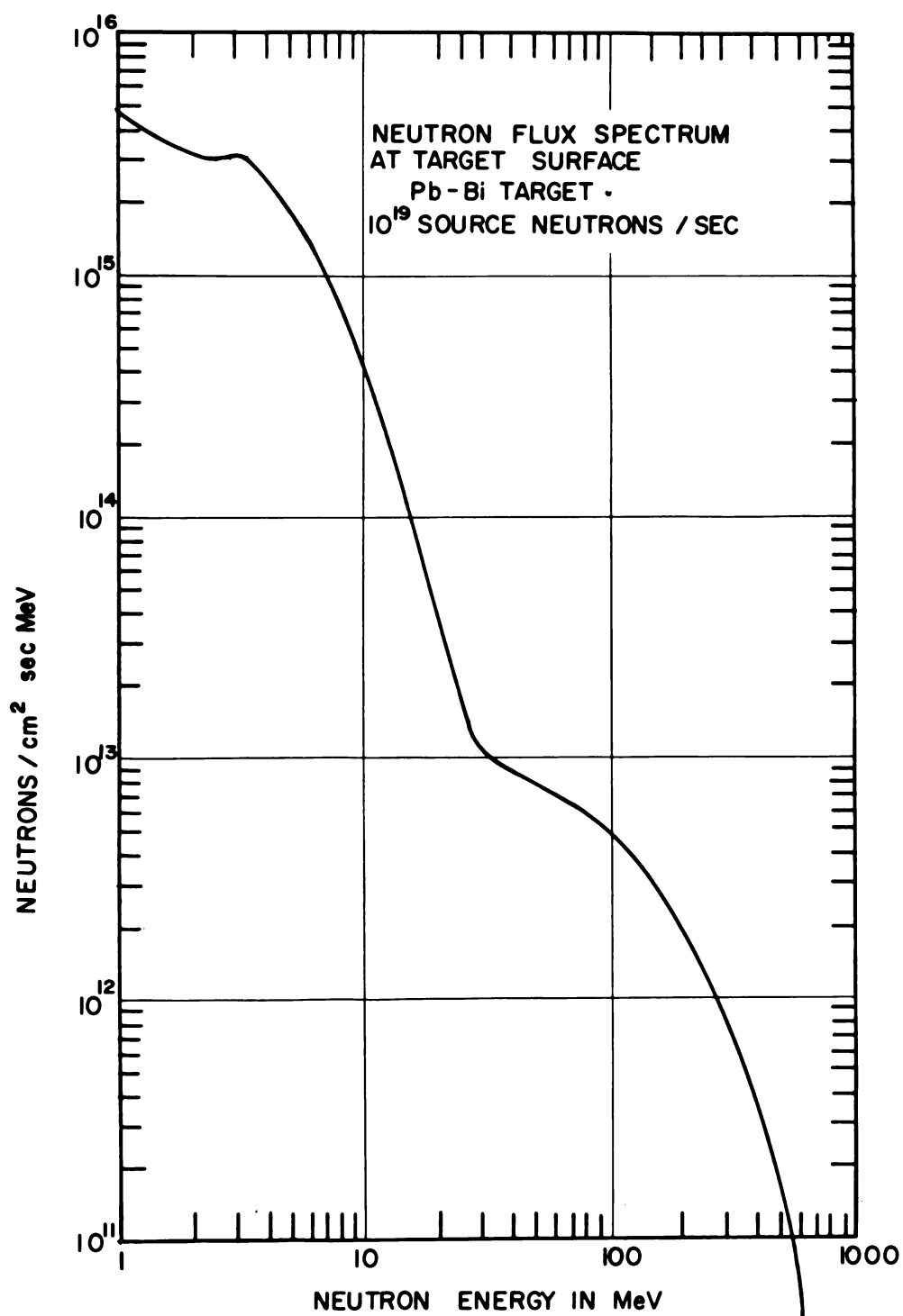


Fig. 8 Fast Neutron Spectrum at Surface of Beryllium

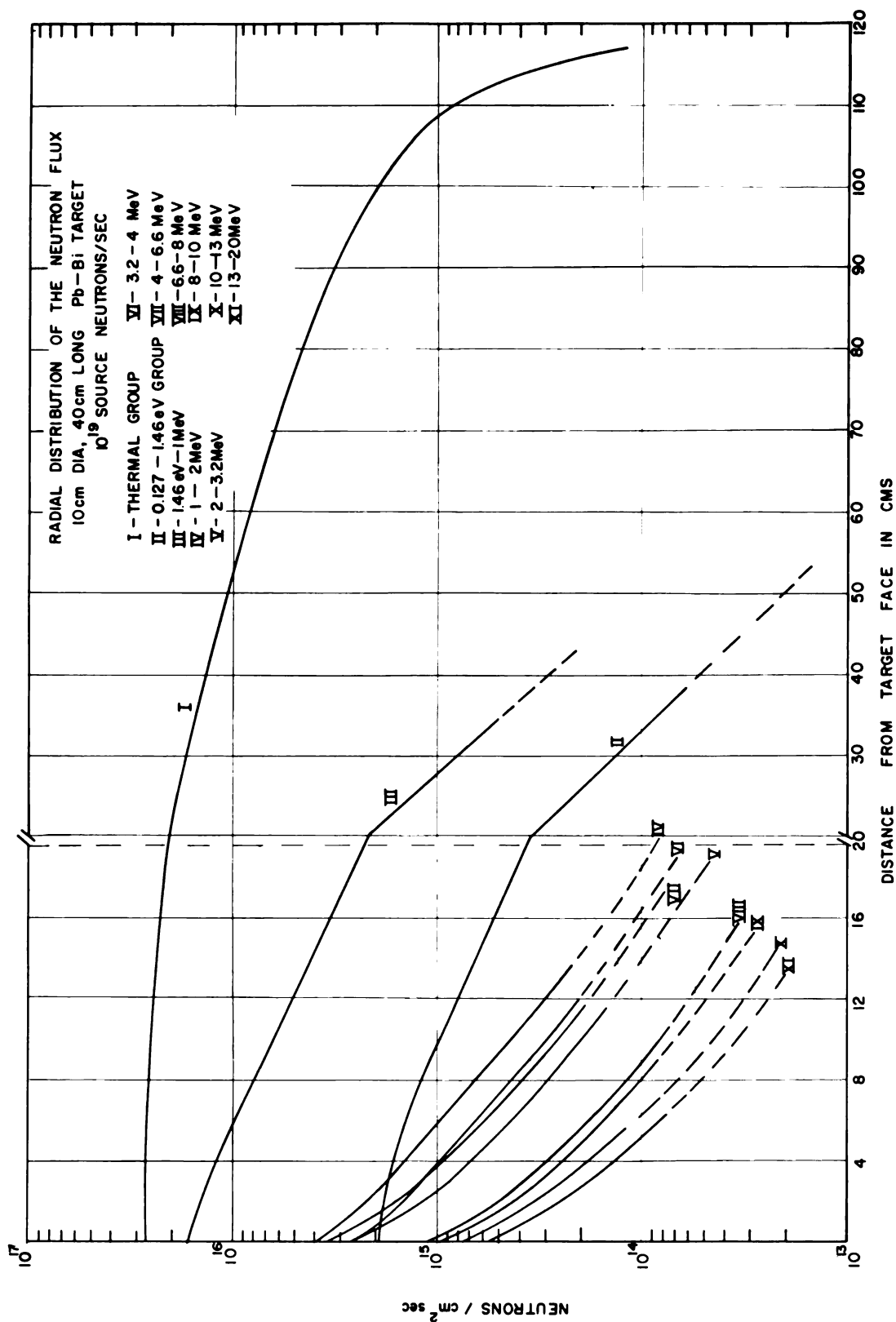


Fig. 9 Radial Distribution of Neutron Flux

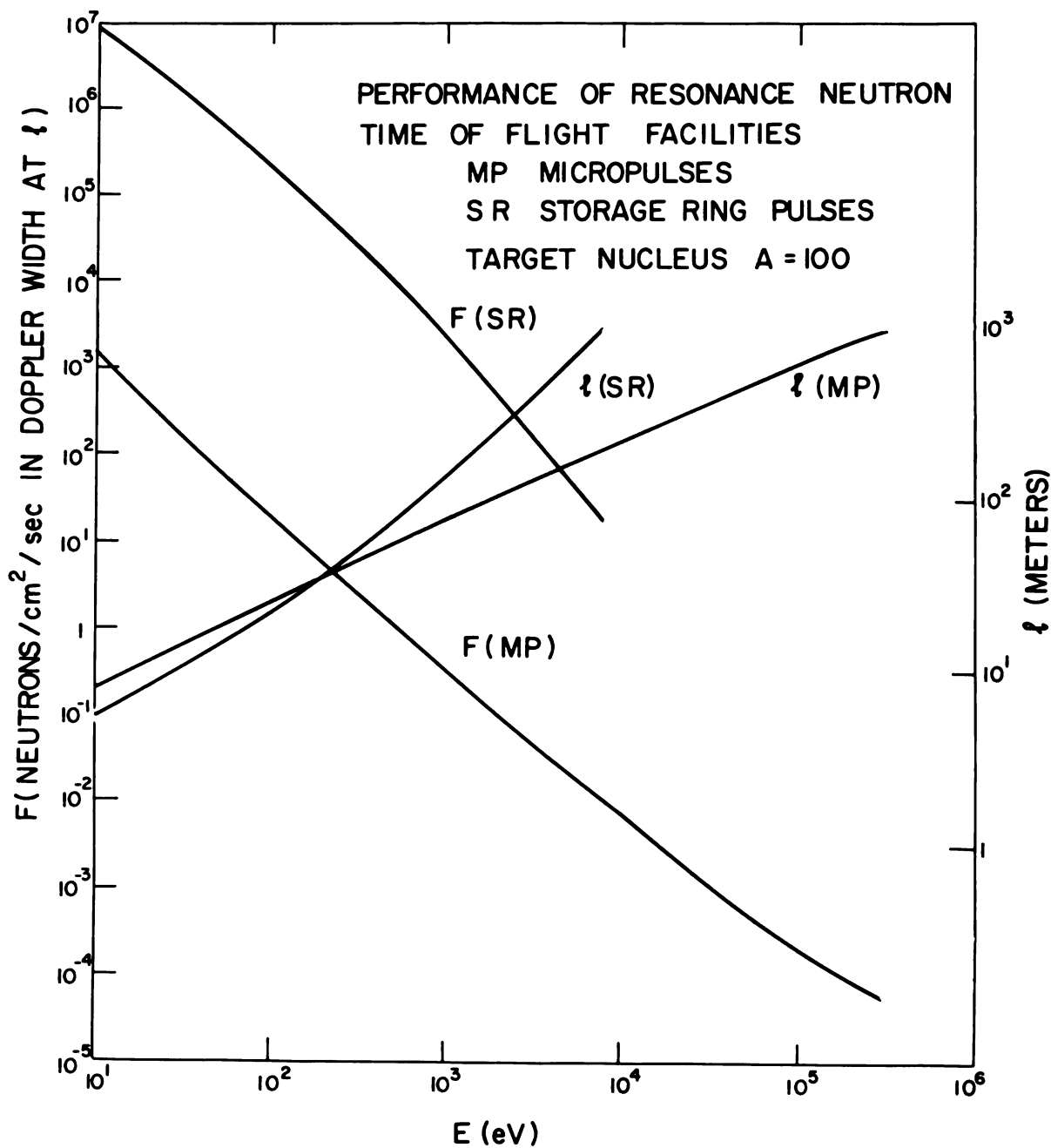


Fig. 10 Performance of Resonance Neutron Time-of-Flight Facilities

DISCUSSION
OF
PAPER V.A. (Bartholomew)

Chairman: L. Rosen

Secretary: D. Cochran

SCHULT: The shielding of high energy neutrons seems to be a major problem at ING. Do you see a possibility to facilitate this through the choice of a good geometry? I suppose you have studied the angular distribution of the fast neutrons emitted from the target in some detail.

BARTHOLOMEW: There certainly would be some gain in looking in the backward direction but perhaps not a great deal. This really has not been studied in as much detail as we would like. The very fast neutrons, as I mentioned, probably would not be a serious problem for many experiments, but they would certainly be a health problem. If one wanted a better thermal-to-epithermal ratio, one could adjust or move the beam tubes out or shorten the beam tubes until one was farther out in the moderator.

SHULT: Do you have a figure on the γ -ray heating (W/g) at a point which is located at, say, 50 cm from the target face?

BARTHOLOMEW: The γ -ray heating has not been calculated. We do not know very much about the γ -ray spectrum. We know that the γ rays from the spallation products would give less trouble than the γ rays from a fission source.

SCHULT: Well, I think it would really be nice if you could measure this. If a fission-type neutron source could be extended so that it would also give something like 10^{16} n/cm²/sec, then the γ heating would be a considerable problem. If ING would be much below a fission source in γ flux, it would not only be a better neutron source because of the improvement which we expect from the increase of neutron flux, but because of the decrease of γ backgrounds.

BARTHOLOMEW: Certainly all parts in the high flux will have to be cooled by circulating water.

MILTON: It is often mentioned that the angular distribution of the high energy neutrons in this reaction is strongly forward-peaked, but what strongly forward-peaked means usually is about a factor of 10. If you look at the amount of shielding that is required, the factor of 10 means something like 18 in. of iron and, if you already have 15 ft. of iron, changing it by 18 in. does not make much difference. You cannot get appreciably closer by looking in the backward direction.

RAIEVSKI: First, I want to say that this is not a criticism of a project of such high interest. My comments concern the production of moderated neutrons, both with regard to the yield and the spectrum. A fission source consisting of a fast plutonium core surrounded by a moderator would produce the same neutron yield as the proposed ING with a power higher by no more than a factor of 2.5. The fission spectrum being softer, the neutron density in the moderator would be higher for the same neutron production. In addition, the very high energy neutrons generated in the spallation process would lead to a poorer signal-to-background ratio, which can rule out the advantage of the higher neutron yield per unit power.

Is it possible to install a cold source in the moderator?

Despite the fact that the figure of 10^{17} n/cm²/sec in pulsed operation with the storage ring is a high number, it is less than can be expected from such a device. This is due to the broadening of the neutron peak in the heavy water moderator. Do you believe it possible to completely or partly change the heavy water to a light water moderator in order to get a sharper thermal neutron pulse and a higher peak value?

BARTHOLOMEW: I am not sure I can remember all the questions. I agree with all the remarks that Raievski made earlier. I think that we have not really studied, or at least I do not recall, the results of a study of the water moderator, but I think certainly one could install this. Our neutron scattering people have not requested cold sources but would perhaps be more interested in hot sources. We have not been thinking about installing a cold source; certainly it would be a heating problem, or a cooling problem!

TUNNICLIFFE: I would just like to make an observation on the light water moderator -- I think the physicists would find themselves in a glorious fight with the commercial products people. The water would capture neutrons the commercial products people would want to use for making cobalt-60. The potential cobalt-60 production is worth quite a lot of money.

MILTON: The difference in the spectrum commented on by Raievski is not really very large. It is true that the spallation spectrum is harder but, on the other hand, you have uranium or plutonium in a reactor. These absorb quite a large number of neutrons, whereas we have bismuth and lead which have about the lowest absorption cross sections known; the spectrum difference is easily taken care of by this difference in absorption in the core, but I do not think that the difference in the hardness of the spectrum is an important point.

RAIEVSKI: I have not quite understood Milton's answer to my question. There were two points in my comment. First, on the neutron yield you mention the difference of absorption between plutonium and uranium in a reactor core and bismuth in a spallation target. This does affect the neutron yield but is taken into account in the factor

2.5 I mentioned. If you look at Spinrad's report you find that you have a figure of 1.9 neutrons available by fission; the factor of 2.5 comes from this figure.

My second remark was on the spectrum. The absorption effect you mentioned in your answer does not influence the spectrum, thus the difficulties connected with the harder spectrum of the spallation process are still present.

BARTHOLOMEW: I think, concerning the spectrum, Milton was referring to the fact that if you look at the thermal flux near the HFBR core, there is a flux depression at the core, then it peaks and then it falls off again with radius. With the ING target, this flux does not have the depression near the core. I think your other number is concerning the plutonium core and we agree with that.

COMMENT FROM BARTHOLOMEW ADDED IN PROOF: Raievski compares neutrons per total target plus moderator power for ING with neutrons per total core power for a plutonium core. A more relevant basis of comparison for ING is neutrons per total target power. With a target power of 38 MW, the efficiency in ING is 28.6 MeV/neutron which is to be compared with ~100 MeV/neutron for the reactor -- a factor of 3.5, not 2.5.

AGERON: I understand from your figures that the fast flux was very large and about 30 times more intense than, for instance, HFBR. Have you studied whether the zirconium container or different targets can withstand such a high fast flux?

BARTHOLOMEW: We do not have the information on radiation damage in the container but we will make arrangements for removing the complete target assembly and the beryllium multiplier as required, probably at rather frequent intervals.

SPINRAD: Do you have any reliable information concerning the buildup of radioactivity in the lead-bismuth source?

BARTHOLOMEW: I think the answer to that is no. Milton may be able to answer that better than I.

MILTON: We do not have any reliable numbers. We have made calculations but these are based on Bertini's Monte Carlo calculations at Oak Ridge which were extrapolated from 400 to 1000 MeV, and so they are not very good. There are no experimental measurements that I know of that help us very much.

BECKURTS: I wonder if you could comment a little bit on the cost of the project?

BARTHOLOMEW: I was asked that question by Livingood at the Gatlinburg Isochronous Cyclotron Conference and then I said that we hoped it would be rather less than twice as much as the LAMPF facility. I think it will be rather more than that. Our present estimate for the basic facility is on the order of \$125 million, but we hope that we can reduce the cost by a lot of study.

ROSEN: Are there other questions or comments? If not, perhaps I would make one. We know from the study at Los Alamos this summer (the study made by F. E. Low and

some of our Los Alamos people in the theoretical division) that ING might be an intense source of K mesons even though it is somewhat below the free nucleon-nucleon threshold; in fact, it will probably make beams 100 times as intense as those that one gets from the Berkeley bevatron. Now, I do not know how you use those in your scheme of things but you ought to be aware of this possibility.

FROM AUDIENCE: How do you produce K mesons below the threshold?

ROSEN: Well, that is a little tricky but it is possible. You use the Fermi momentum of the nucleons in the heavy nucleus. Another way of saying this is that one makes use of the high momentum components of the target nucleus wave function.

PROSPECTS FOR HIGH CURRENT ACCELERATORS*

Darragh E. Nagle, Los Alamos Scientific Laboratory, New Mexico, U.S.A.

ABSTRACT: A number of studies have been made and several projects are underway to construct very high current accelerators in the energy range 200 - 1000 MeV. In this talk we shall discuss some of the problems which appear at higher currents. They include: problems of beam quality from the ion source, preaccelerator, and transport system; problems of space charge defocusing of strong focusing fields; beam loading effects; cavity-beam instabilities or amplification; targeting problems; and the economic and systems problems of supplying the power for the accelerating beam.

1. INTRODUCTION

Suppose that you want to build an accelerator for protons: The energy is to be 1 GeV and the average current greater than 10 mA. What are the principal technical problems and how might they be solved? What technical and economic problems turn up if still higher average beam currents are required? The uses of such a machine as an intense neutron source are discussed elsewhere in this symposium; notice however that since a substantial fraction of the beam power (10 MW or more) ultimately produces nuclear reactions, the problems of induced radioactivity in the accelerator and in the target region are of a magnitude more frequently encountered in reactor design than in accelerator design. For example, a beam loss of 50% during acceleration is acceptable in present day synchrocyclotrons where the average current is a microampere; if the beam accelerated is to be 10 milliamps we should design to lose less than a few tenths of a percent of the beam. The accelerator, in spite of careful design, may still lose beam and become radioactive. Therefore, components which may become active should be easily and rapidly serviced; in some areas by the operating staff, but certainly in the target region by remote manipulation. The problem of supercritical excursions for reactors has perhaps an analog for accelerators; if the beam, because of component failure or any other reason, strikes the structure of the machine, a burst of intense radiation is produced and a part of the structure will be vaporized. This means that a fast shut-down system is required. Finally, the costs of providing the rf power and building the

structure must be assessed. I will discuss briefly the technical implications of the above criteria, and will to some extent emphasize the linear accelerator because I know it better. To a considerable extent the different types of accelerators, i.e., linac, cyclotron, SOC, etc., have these problems in common. The SOC accelerator, a contender for the intense neutron generator can be discussed better by Russell and by Bartholomew in these seminars. This talk is not aimed at accelerator specialists; therefore, I have attempted to define all the jargon.

2. BEAM QUALITY AND INTENSITY

Possible ways of keeping the beam from hitting the walls are to use very large apertures, to use strong restoring forces, or to maintain the best beam quality and accelerator tolerances throughout. The large aperture approach was used first, chronologically, but becomes uneconomic for high power, medium-energy accelerators. Most modern accelerators are of the strong focusing type and usually have several sections. The trend is to improve the beam quality throughout the accelerator by starting at the injector with a high quality beam and maintaining the quality in the sections following. The quality of the beam refers to the reciprocal phase space volume occupied by points representing the particles:

$$Q^{-1} = V = \int dp_1 dq_1 dp_2 dq_2 dp_3 dq_3$$

Liouville's theorem says that if the forces derive from a Hamiltonian, $H(p,q,t)$, the representative points move in phase space so as to conserve V . However, if we add random forces (voltage fluctuations for example) V will tend to grow. Also, while some types of systems may conserve V , they distort the boundary surface so that the practically useful volume V has increased. To preserve quality, errors must be kept small and transformations avoided which badly distort the phase space contours. The phase volume of the beam out of part of the accelerator is called the emittance; the volume of phase space accepted is called the acceptance.

In general Liouville's theorem would hold separately for a pair of canonically conjugate variables: $\int dp dq = \text{const.}$ The variables energy and phase (E, φ) or momentum and displacement ($m\dot{r}, r$) are not strictly conjugate because of coupling terms in the Hamiltonian; nevertheless, one customarily talks of the $E\varphi$ admittance and acceptance and of the transverse (p_r, r) admittance and acceptance, and often they prove to be approximately constant. The normalized transverse emittance is defined as

TABLE 1
Evaluation of High Intensity Ion Sources

Reported maximum	Units	With large plasma expansion and plasma boundary focusing				
		PIG Source	RF Source	Duo-plasmatron Source	Lamb-Loefgren "Magnetic" Source	Modified Duoplasmatron Source With Hot Cathode
Ion currents: after single electrode acceleration	mA	150	350	1000	1000	800
after C-W acceleration	mA	90	250	150	—	400
Physical characteristics: beam energy spread	ev	<10	10	<1	—	—
typ.max.proton perc.	%	50-80	60-90	60-90	95	—
Characteristics: related to complexity of auxiliary equipment						
gas consumption	Ncm^3/h	10-100	10-100	50	2000	100-200
(power input)*	kw	0.5-5	1-10	0.5-2	5-10	1-5
duty factor)						
power for solenoids, etc.	kw	1	0.5	0.5	10	0.5
cathode heating	kw	—	—	0.1-0.5	1	0.1-0.5
discharge current	A	1.5	—	10-40	100	20-100

* Related to source discharge current only.

TABLE 2

Ion Source Characteristics

Source	V_{dc} MeV	$\eta = \sqrt{\frac{2V_{dc}}{940}}$	E cm mr	E η cm mr	I A	I_{max} A	$\frac{I}{(E\eta)^2}$ Acm ⁻²	Reference
P.I.G. + Col.	0.75	0.04	8	0.32	.032	.04	0.40	Van Steenberg
Duopl. + Col.	0.75	0.04	9	0.36	.057	.08	0.44	Van Steenberg
Cyclo. Chimney Arc	0.03	0.008	13	0.10	.003	.004	0.30	Blosser and Mallory
Duopl. + Short	0.50	0.0326	8.6 π	0.28 π	>.250	.700	>0.4	Hugenin
Duopl. + Expansion Cup	0.028	0.0077	0.041	0.31	.080	—	0.8	H. Wroe, ENL-HW-2

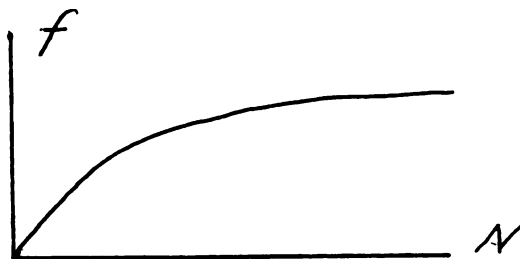
$$N = \frac{p_z}{mc} \int \frac{dp_r}{p_z} dr = \eta \int d\theta dr \quad \text{where}$$

$$\eta = \beta(1 - \beta^2)^{-1/2}, \quad \beta = \frac{v}{c} \quad \text{and} \quad \vartheta = \frac{p_r}{p_z} < 1.$$

ϑ is the divergence angle of the particle with respect to the z axis. We expect N to increase rather slowly with energy in a well designed system. The problem of designing the accelerator so as not to lose beam becomes one of matching the emittance of one section to the acceptance of the next. Therefore, we discuss the intensity, emittance, and acceptances of various portions of an accelerator and how various effects change them.

3. ION SOURCES, DC EXTRACTORS, AND ACCELERATING COLUMNS

The ion source, the ion extraction system, and the accelerating column tend to be designed and tested as a unit. The diagram shows a typical ion source and column characteristic, wherein f , the fraction of the total current within an emittance N , is plotted versus the emittance.



The brightness B , defined as the ratio of f to N^2 , approaches a constant for small f in most systems. Van Steenberg has summarized the state of ion source development.¹ Table 1 is reproduced from his paper. Table 2 compares the emittance and the brightness of ion sources. Van Steenberg suggests that in principle an order of magnitude improvement in ion source brightness may be possible. References for column design may be found in Pierce² and Curtis.³

The brightness attained in duoplasmatrons is higher than for P.I.G. sources, RF sources, or the chimney arc type ion sources used in cyclotrons. For example, Mallory and Blosser⁴ and Mallory⁵ report a normalized emittance of 0.10 cm-mrad for 30 keV and 3.5 mA, which is about a third the normalized brightness of the duoplasmatron (see Table II).

We conclude that ion source and column development is progressing rapidly; the present sources are adequate for linacs in the 100-200 mA range.

4. BEAM TRANSPORT

Here one is concerned with providing magnetic focusing to contain the beam against electrostatic effects of space charge: this problem was treated by Walsh⁶ and by Kapchinskij and Vladimirkisij.⁷ Under the assumption that the particles start on the surface of an ellipsoid in phase space, it can be shown that they will remain on such an ellipsoid under the action of space charge and the lenses of a beam transport system, i.e., beam quality is undisturbed. R. Emigh⁸ has estimated that space charge effects in a 44 ft long 750 keV beam transport system with four lenses would in practice add no more than 0.04π mr-cm to the normalized emittance.

5. BUNCHER

It is usually necessary to bunch the beam before it enters the linac. The buncher impresses a velocity modulation $\Delta v \cos \omega t$ on the beam. After a drift length L the arrival times of the particles at the linac cluster according to the relation

$$t_{\text{linac}} = t_{\text{buncher}} + \frac{L}{v_0 + \Delta v \cos \omega t}.$$

The buncher design involves a compromise between the induced energy spread $\Delta E/E = 2\Delta v/v_0$ and the drift length L required according to the law $\frac{\Delta E}{E} \left(\omega \frac{L}{v_0} \right) \sim 1$. (Increasing L decreases the energy spread ΔE but allows more time for space charge defocusing to operate.) ΔE must be large compared to any ripple or noise on the beam out of the column, a severe limitation on multistage bunchers.

6. ALVAREZ SECTION RADIAL MOTION

An Alvarez linac consists of a series of 2π mode sections or tanks typically at 200 MHz. This will constitute the first 100 MeV of our linac.

The radial defocusing of the rf fields and of space charge forces is overcome by quadrupole magnetic lenses located in the drift tubes. For the LASL linac the two forces would be of equal magnitude for I_{beam} of several hundred mA. For currents of this magnitude the quad currents must be increased over the settings for a zero intensity beam. Since the quad currents are already high, this problem must be taken into account in going to higher currents. The group at CERN⁹ have noted the following changes in quad settings as the beam has improved over the years:

Beam Current mA	Tank I Quad Currents (arbitrary unit)
10	0.75
21	0.9
50-65	0.95-1.05

This experience is consistent with the above statements. Studies with computers of the radial motion from 50 to 100 MeV show the radial emittance staying about constant even in the presence of reasonable quadrupole errors. The normalized radial emittance at 100 MeV with errors and coupling effects is about 3.6 mr cm. The normalized radial acceptance of the waveguide linac at 100 MeV is typically about 10 mr cm depending, of course, on the apertures used.

Table 3 summarizes our discussion of radial matching. Although the match is good, we should emphasize that beam quality worsens in a multiplicative rather than an additive way and it would be desirable to use less than the full acceptances. In this way we could get a further improvement in beam quality.

7. LONGITUDINAL MOTION AND THE TRANSITION ENERGY

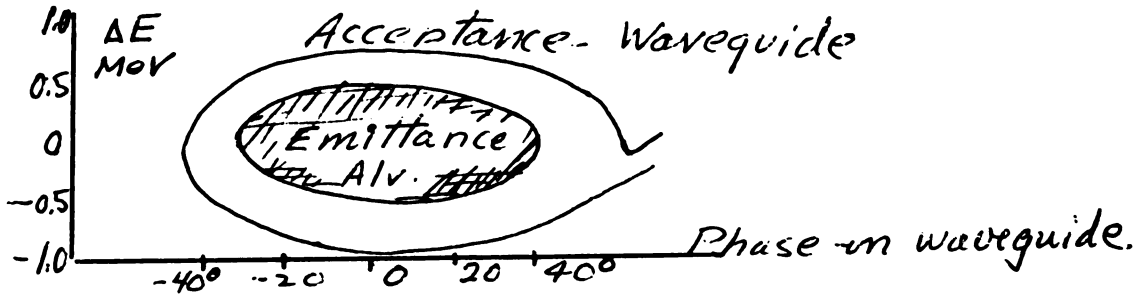
In the LASL linac, for reasons of power economy, a transition is made at 100 MeV to a higher frequency (800 Mc) (harmonic number of 4) and to a waveguide structure rather than an Alvarez structure. The longitudinal phase match at the transition energy of 100 MeV between the Alvarez and waveguide portions is adequate but not generous. The diagram shows the

TABLE 3

Representative Values of the Normalized Radial
Acceptance and Emittance of Several Portions of
a Linear Accelerator

Section	Energy MeV	Acceptance mr-cm	Emittance mr-cm
Ion Source + Column	.750		0.25
Beam Transport	.750		(+.13)
Alvarez	.75	2	
Alvarez	100		3.6
Waveguide	100	10	
Waveguide	800	10	

acceptance region of the waveguide portion and the emittance region of the Alvarez part.



The matching problem enters into the high current regime considerations because the space charge reduces the longitudinal damping. For $v < c$ and for a peak rf field $E_z > \frac{dW}{dz} \frac{s}{E_z}$, the rate of energy gain for the synchronous particle, there exists a synchronous phase $\varphi_s = \cos^{-1}(dW/dz/E_z)$ in whose neighborhood longitudinal oscillations of the particles are stable. For a deviation φ away from the synchronous phase the restoring force is

$$\approx \{eE_z \sin \varphi_s - eE_z (s.c.)\} \varphi.$$

In the first Alvarez gap the second term is about 10% of the first. This is okay, but at 300 or 400 mA (depending on the bunch width) the longitudinal focusing would vanish. One may raise E_z and improve the phase stability, but then transverse stability requires still higher quad currents.

8. INSTABILITIES, RESONANCES, AND BEAM BLOWUP

We may define machine resonances by the equation

$$\sum_{i=0}^n a_i \omega_i = 0$$

where ω_0 is the frequency of revolution in a cyclic machine ($\omega_0 = 0$ for a linac or a SOC), and $\omega_1 \dots \omega$ are the frequencies of oscillation in the degrees of freedom of the single particle motion. The a 's are small whole numbers. One chooses the operating point of the machine so the left-hand side of the equation doesn't vanish. The ω 's are functions of the external magnetic and electric fields, and in general are functions of the beam current. Thus a large current may move the operating point onto a resonance. In circular machines this has been a limitation. Dzepolov¹⁰ estimates a shift in the axial oscillation frequency of a weak focusing relativistic cyclotron of 100% would be caused by a current of 13 mA.

For linacs it is relatively easy to avoid the serious resonances: the principal one is $2\omega(\text{transverse}) = \omega(\text{phase})$ which is missed by properly arranging the magnetic focusing periodicity. Space charge would reduce the

damping of the phase oscillations; this must be considered if a frequency transition is to be made.

Some other instabilities in linacs and other machines are a result of coherent interactions between the beam current and parasitic rf fields. An example is the backward wave deflecting mode instability.¹¹ The starting current for this instability may be a few hundred mA for electron linacs and for the LASL linac 6 or 7 A. A more subtle and potentially more serious interaction is the deflecting mode interaction recently observed at SLAC:¹² at a critical current of about 20 mA peak, depending on parameters of machine operation, the beam begins to break up, starting at the end of the pulse. The breakup phenomenon is accompanied by the appearance of a TM₁₁-like mode at 4140 MHz radial oscillation of the beam. Although not all features of the phenomenon have been studied, it appears that the TM₁₁ mode oscillation is excited by the coupling between the bunched deflected beam and the TM₁₁ deflecting band. Information is carried by the beam deflection from tank to tank, so that the whole accelerator acts like a sort of distributed amplifier for this frequency. In the case of the proton high current linac of the LASL design the TM₁₁₀ deflecting mode band lies around 1600 MHz, is a backward wave, and the bandwidth is about 1%. Preliminary experiments with electron beams in the LASL waveguide tanks indicate some coupling of this mode to the beam. However it appears that it will be possible to make the starting current for breakup large compared to the 17 mA required in the LASL design.¹³ This problem must be considered carefully, but probably will not prevent the attainment of the 65 mA proposed by Chalk River.

A related instability for circular accelerators is the resistive wall instability, which arises from a coherent interaction between the beam and the walls of the vacuum chamber.¹⁴

9. ACCELERATING STRUCTURES

Here at LASL we believe we have made a contribution to the push to higher beam currents in our development of a waveguide accelerating structure of high efficiency, excellent power transfer characteristics, good voltage holding, and reasonable machining tolerances.^{15,16} All tests of this structure, a so-called $\frac{\pi}{2}$ - mode resonantly side-coupled waveguide structure, have been satisfactory.

10. RF POWER SYSTEMS

If we consider a 0.8 GeV, 65 mA av current accelerator, the beam power is 52 MW. The power required to make up copper losses in the tanks will be about the same, so a 100 MW rf power system is required, which will be the most

expensive part of the accelerator. It must have a low fault rate, since all the units must be on if the machine is to function at full energy.

The rf system proposed for the IASL linac¹⁷⁻¹⁹ might serve as a starting point for the design of a linac with a more intense beam. Most of the design features lend themselves to scaling up the beam intensity. One open question is the relative merits of the several kinds of 800 MHz tubes which have been proposed for the final amplifier, namely, triode, klystron, and crossed field amplifier.²⁰⁻²³ At the present time, our own tests do not indicate a manifest superiority of any one type of 800 MHz amplifier. At 200 MHz the RCA 7835 triode is chosen as the final power amplifier.

11. TARGETING

Target design is a major problem already for the meson factory, where we have been planning for the relatively modest power levels of 800 KW. Heat removal, radiation damage, and radioactivity production must be taken into account in target design. We expect to have several targets in the beam. The meson producing targets would be 5 KW to 100 KW units. The remainder of the beam energy would go to a so-called dump, which could be used for neutron production, neutrino production, and for experiments on nuclear activation, radiation damage, etc.

The two following tables list some advantages and disadvantages of target materials for our service, and give the characteristics of some graphite target mockups which in their initial tests seemed promising for use as pion producers.

Remote handling of targets and equipment in the target room is essential. Remote handling of most components of the IASL accelerator will not be required.

If the beam power is raised a hundredfold, target design will be much more difficult, and remote handling of the accelerator parts may be necessary.

12. CONCLUSION

No insurmountable problems are evident at present for designing accelerators of currents of 50 mA or so, such as are now being contemplated for use as intense neutron sources. Any such accelerator will be very expensive and will require a vigorous development, especially of the rf system and of the targeting. Very careful design of the accelerator sections to avoid loss of beam will be essential.

TABLE 4

Stationary Targets Under Consideration
For Pion Production

<u>Material</u>	<u>Coolant</u>	<u>Difficulties or Disadvantages</u>
Graphite (coated)	Water	Physical Properties
Beryllium (coated)	Water	Physical Properties
Be O (coated)	Water	Physical Properties
Water	Self	Coolant Activation, Beam-Vacuum Interface
Liquid Lithium	Self	Coolant Containment, Beam-Vacuum Interface
Be, H ₂ O Laminate	Self	Coolant Activation, Beam-Vacuum Interface
Copper	Water	Lower Production Efficiency, Higher Neutron Background Compared to C or Be

All targets may have problems due to thermal cycling and radiation damage.

TABLE 5

Mock-up Target Tests
A.C. Electric Heating

	<u>No. 25</u>	<u>No. 29</u>
Material	Graphite	Graphite
Dimensions	10.8 cm x 1 cm diam	4.4 cm x 1/2 cm diam
Coolant	Water	Water
Flow (paraxial)	23 gpm	13 gpm
Pressure Drop	~ 75 psi	~ 143 psi
Max. Dissipation	94 kW	38 kW
"Indefinite" Life		
Dissipation (I L D)	62 kW	24 kW
Avg. Power Density (I L D)	7.3 kW/cm ³	6.9 kW/cm ³
Surface Heat Transfer Rate (I L D)	1.86 kW/cm ²	3.44 kW/cm ²
Expected Average		
Proton Beam Dissipation	45 kW	20 kW

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* Research supported by the U. S. Atomic Energy Commission.

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DISCUSSION
OF
PAPER V.B. (Nagle)

Chairman: L. Rosen
Secretary: D. Cochran

TUNNICLIFFE: I have a question on your calculation of the losses in the structure which you give as about 50 MW peak power. I take it that is the figure you calculate for your machine.

NAGLE: I was calculating for your current (65 mA) but for our gradient and final energy. Our peak power is about 45 MW total.

TUNNICLIFFE: To reduce our gradient, you see, we double the length of the machine and the losses are about half.

NAGLE: Well, I have done the scaling wrong because I took our gradient.

MILTON: Did I infer correctly from your last figure that the water target was introduced by a windowless vortex action? I could not quite understand the diagram.

COMMENT ADDED IN PROOF BY NAGLE: The actual target is a solid graphite cylinder. The cooling water flows in a helical path about the graphite. The target has no window in the direct beam and the direct beam sees only graphite. The scattered pions outside the forward cone do go through the water and through a metallic wall separating the water and vacuum.

SESSION VI

CONTRIBUTED PAPERS

APPLICATION OF PULSED NEUTRON SOURCES
TO STRUCTURE AND LATTICE DYNAMICS
STUDIES OF SOLIDS*

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and

Institute of Experimental Physics, University of Warsaw, Poland

ABSTRACT: The actual and potential applications of pulsed neutron sources to studies of structure and lattice dynamics of solids are discussed.

A brief description of the time-of-flight (TOF) method for structure analysis is given; some typical results obtained both for powders and single crystals are presented; and a comparison between the conventional and TOF methods is made. Special emphasis is put on resolution and intensity problems. It is shown that for magnetic structure analysis and small lattice distortions studies the TOF method used with repetitively-pulsed reactors, e.g., the IBR reactor at the Joint Institute of Nuclear Research at Dubna, USSR, is preferable. Special features of the TOF method and their nonconventional applications to structure studies are discussed, e.g., studies of the influence of very high repetitively-pulsed external fields on structure and studies of transient phenomena.

A new modification of the TOF method for lattice dynamics research is briefly presented.

A method for simultaneous structure and lattice dynamics studies is presented. This may be useful in studies of the relation between the structure of a solid and its lattice dynamics, especially in studies of phase transitions and the influence of external fields both on structure and lattice dynamics.

The final conclusion, that high intensity pulsed neutron sources open a new promising area of research on structure and dynamics of solids, is emphasized.

* Part of the work, described in this paper, was performed at Świerk, Poland, under research contract No. 255/RB of the International Atomic Energy Agency.

1. INTRODUCTION

The study of lattice dynamics of solids is performed nowadays by applying both neutron crystal and TOF spectrometers (1). In the case of crystal spectrometers, nuclear reactors with a steady (in time) flux are used. The TOF method was also applied at steady flux reactors and the neutron pulses were obtained by means of choppers. In 1961 the first repetitively pulsed reactor IBR at the Joint Institute of Nuclear Research at Dubna, USSR (2), was put into operation. At the beginning it was meant mainly for nuclear physics research, but very soon it was applied also to the study of lattice dynamics (incoherent inelastic scattering) using the TOF technique (3,4) imposed by the pulsed feature of the neutron source. In the course of time this method has been improved by using a beryllium filter in front of the counter (5), and in a recent paper (6) the use of a beryllium filter and a crystal monochromator in front of the counter was reported. In the meantime some studies on dispersion relations in solids (coherent inelastic scattering) using the TOF method have been also started at the IBR reactor (7,8).

Structure analysis of single and powdered crystals was performed until recently using exclusively conventional neutron crystal spectrometers. Although the TOF method of crystal structure analysis was suggested some time ago by Egelstaff (9) and Ringo (10) and maybe several other physicists had it in mind, no experimental work on this method has been published, except the observation of higher order reflections from single crystals by Spencer et al. (11) and Whittemore (12) and Lowde's paper on a new rationale of structure factors measurements in single crystals (13). Recently this situation has changed considerably.

Starting in 1963 Buras et al. (14-17) developed, discussed and checked experimentally the TOF method for structure analysis of crystal powders at the EWA reactor at Świerk, Poland. About the same time this method was further developed at the pulsed reactor IBR at the Joint Institute for Nuclear Research at Dubna (17-22), where later it was also successfully applied to the crystal and magnetic structure analysis of powdered BiFeO_3 (23-25). Similar studies on powders using the TOF method have been undertaken at the Danish AEC Research Establishment, Risø (26), AWRE, Aldermaston, UK (27), and at the Northwestern University at Evanston, Illinois, USA, in collaboration with Argonne National Laboratory, USA (28). Brugger et al. used the TOF method at the National Reactor Testing Station, Idaho Falls, USA, for neutron diffraction studies of samples at high pressure (29). Shull discussed this method at a Symposium on pulsed intensity fission neutron sources held in Washington in 1965 (30). It is also briefly described in the book of Izumov and Ozerov (31).

Experiments performed at Risø and Świerk have demonstrated the usefulness of the TOF method for single crystal studies (32) and at the VII International Congress of Crystallography in Moscow (1966) Lebech and Mikke reported experiments on antiferromagnetism in single crystals of chromium-rhenium alloys performed using the TOF method (33).

Recently Buras discussed the possibility of the simultaneous study--using one sample--of structure and lattice dynamics of solids by applying the TOF technique (34,35).

The aim of this paper is to present the actual and potential applications of pulsed neutron sources to studies of structure and lattice dynamics of solids. It includes a large part of the material already presented (35) with some changes and additions. The information is also brought up to date.

2. STRUCTURE STUDIES

2.1. General description of the TOF method for structure analysis and typical results

2.1.1. Powdered crystals

Figure 1 illustrates the method. A pulsed neutron beam is scattered on a powdered crystal and the intensity of the scattered neutrons at a fixed angle $2\theta_0$ is measured by means of neutron counters connected to a multichannel time analyser. As a result an intensity vs wavelength curve is obtained. Figure 2 shows a neutron pattern of powdered silicon obtained at Swierk (16), and Fig. 3, a neutron pattern obtained at Risø for PbTiO_3 (26). Note the good resolution which enables the (100) and (001) peaks to be separated.

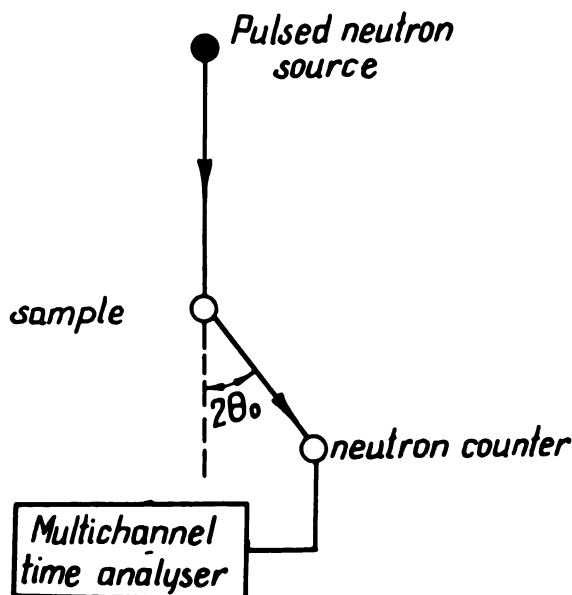


Fig. 1 The TOF arrangement for structure analysis of powdered samples

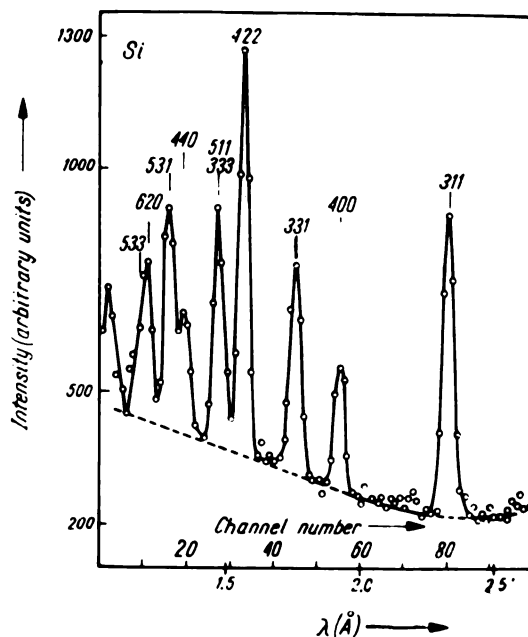


Fig. 2 Diffraction pattern of powdered silicon obtained at the BWA reactor at Swierk using a 100 channel time analyser (16)

The peaks are indexed in the usual manner using the Bragg equation, and the relative values of the structure factors are obtained by applying the formula for integrated intensity derived for this method by Buras (15).

Figure 4 shows the arrangement used at the IBR reactor at the Joint Institute for Nuclear Research at Dubna (21). Scattering angles of 52° and 90° were used and in addition a special collimator (16,21) was employed, which enabled neutrons to be collected from about $1/6$ of the Debye-Scherrer ring (for a scattering angle of 90°). The intensity was measured by means of scintillation counters with enriched B^{10} (36). The fast neutrons are slowed down by several cm of water (or water with a small addition of H_3BO_3) located close to the reactor. The flight path was about 18 m, and the average reactor power several kW. Figures 5-7 show some typical results obtained for silicon, aluminum and $BiFeO_3$ powders, respectively (21,24). Note on Fig. 7 the very good resolution demonstrated by the splitting of peaks (111) and ($\bar{1}\bar{1}\bar{1}$) due to a small rhombohedral distortion ($89^\circ 36'$) of the $BiFeO_3$ cubic lattice (24).

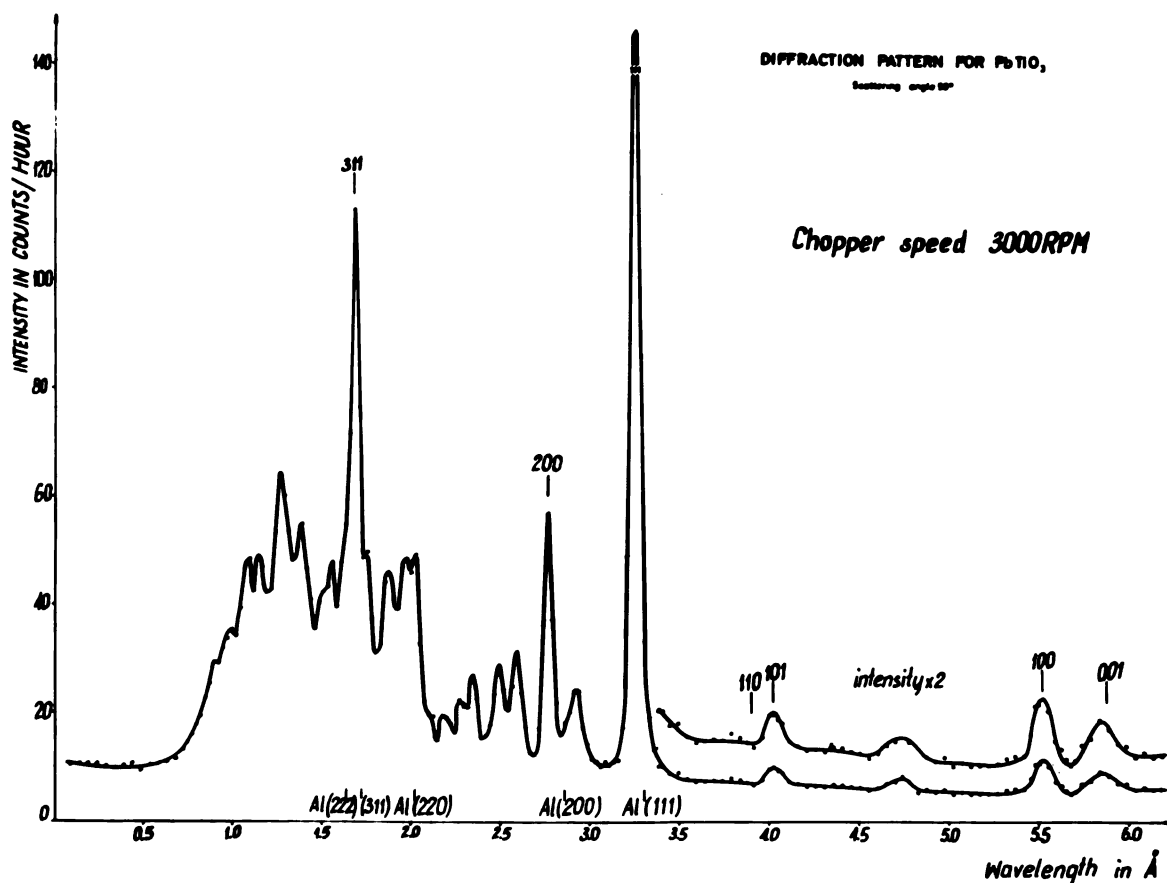


Fig. 3 Diffraction pattern of powdered $PbTiO_3$ obtained at the DR 3 reactor at Risø (26)

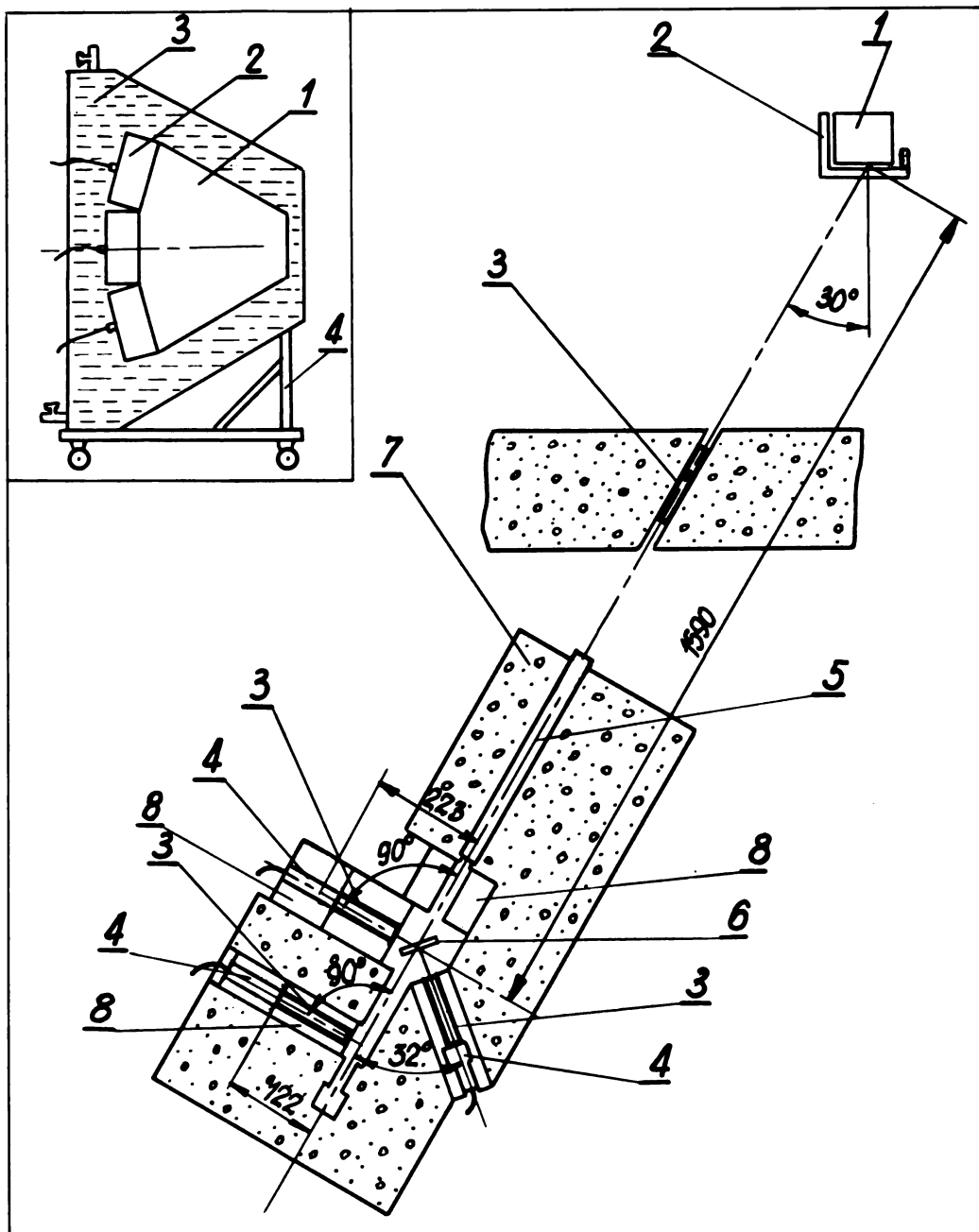
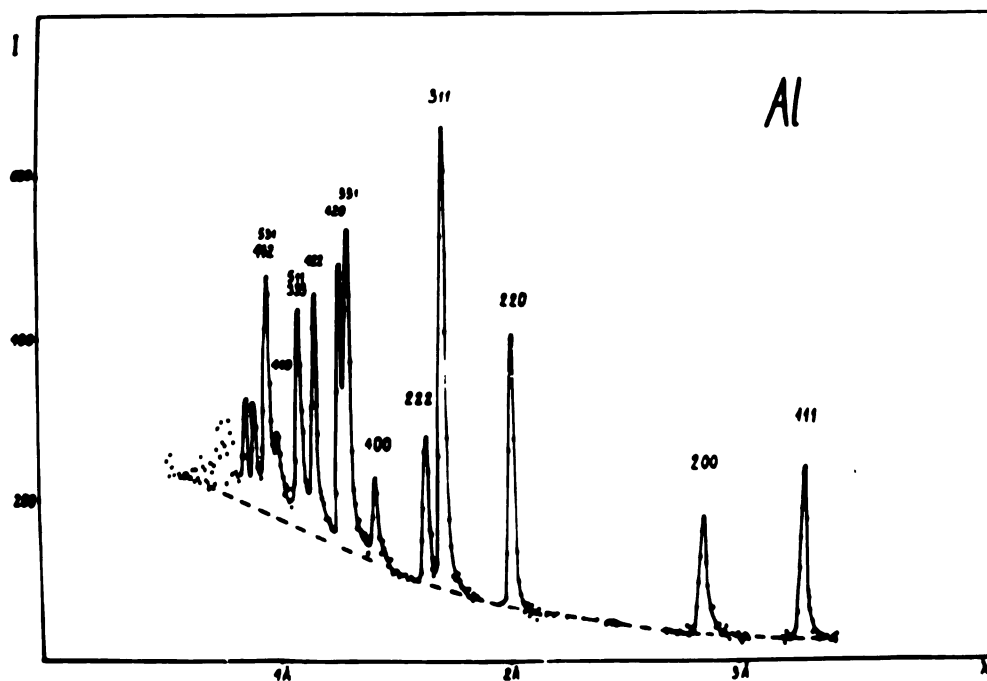
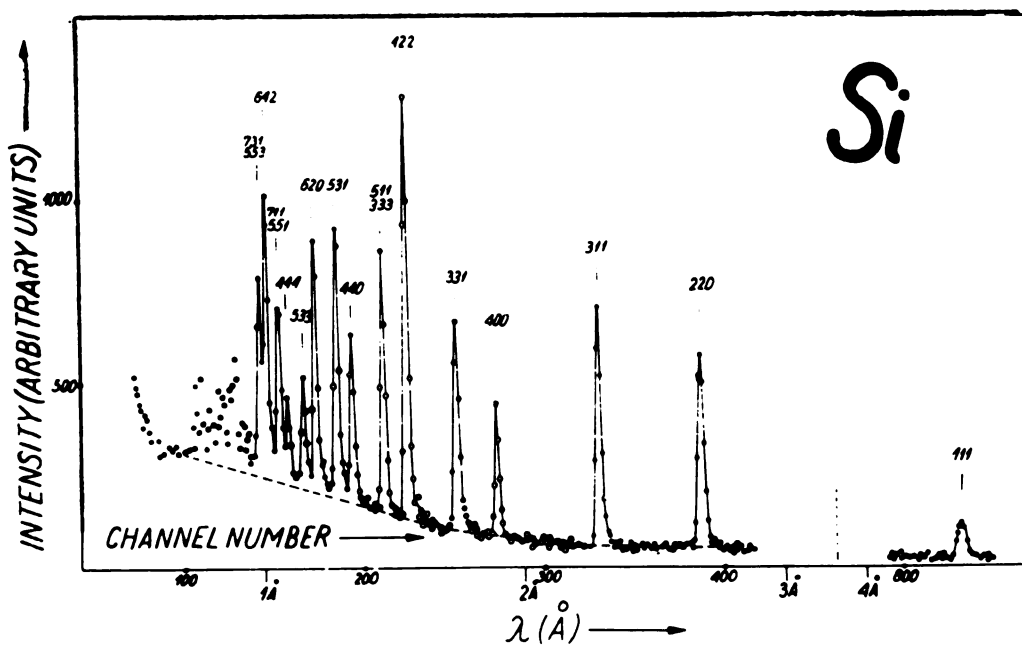
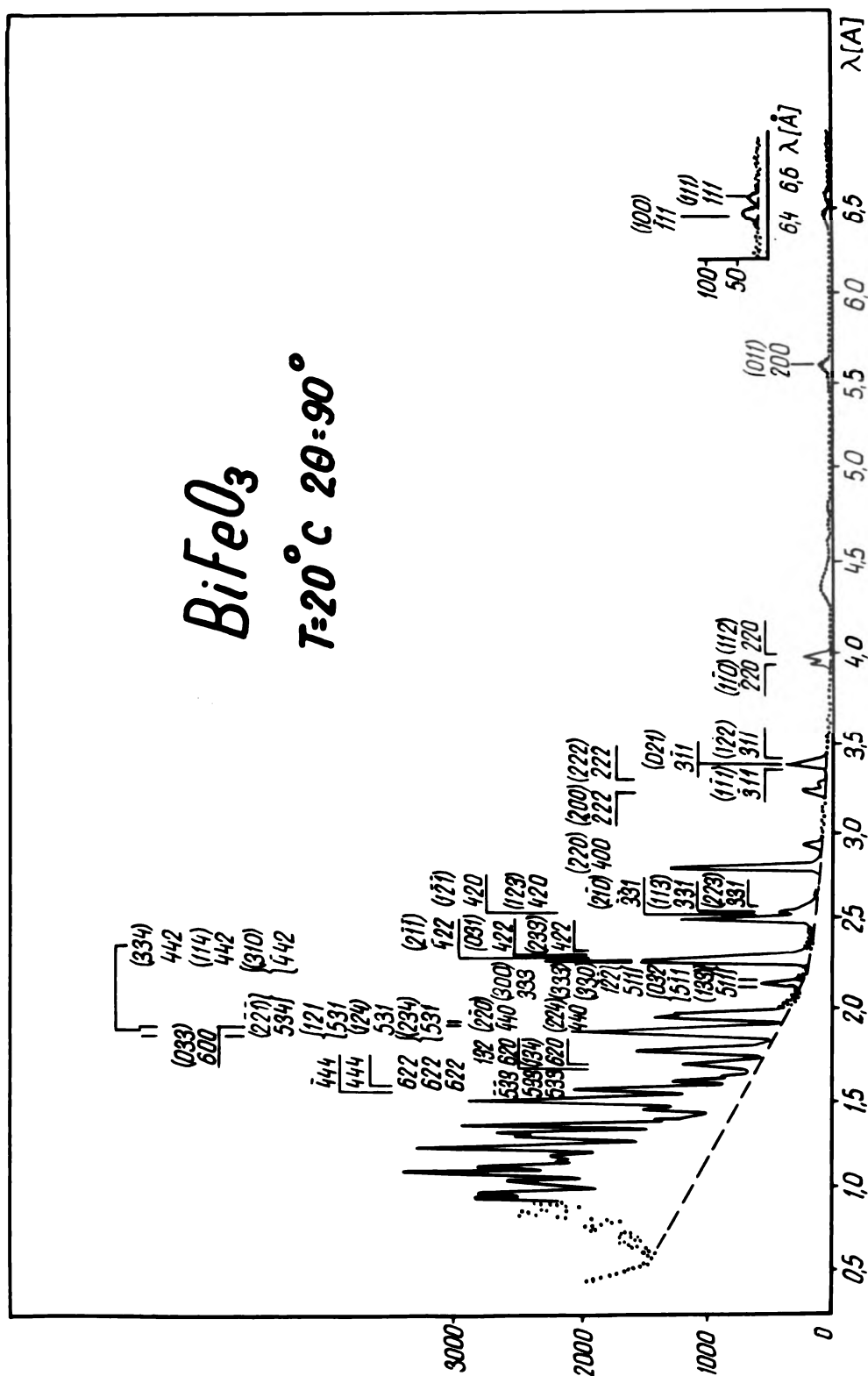


Fig. 4 The experimental arrangement used at the IBR reactor at the Joint Institute of Nuclear Research at Dubna;

Main figure: 1 - reactor core, 2 - moderator, 3 - collimators, 4 - scintillation counters, 5 - neutron (vacuum) guide tube, 6 - sample, 7 - concrete shielding, 8 - water shielding.

Inserted figure - special type collimator to collect neutrons from about 1/6 of the Debye-Scherrer ring: 1 - Soller collimator, 2 - scintillation counters, 3 - water shielding, 4 - base.





2.1.2. Single crystals

Figure 8 illustrates the principle of the method developed at Risø and Świerk (32). A pulsed collimated polychromatic neutron beam is scattered on a single crystal sample with its zone axis perpendicular to the plane of the figure. Their intensities are measured by a circular type detector connected to a multichannel analyser, which segregates the reflection (peaks) according to the reflected wavelengths. The sequence of peaks in this method does not follow the simple rules as in the case of powders, however, several methods for indexing can be used. The computation of the observed structure factors can be carried out using Laue's formula for the integrated intensity.

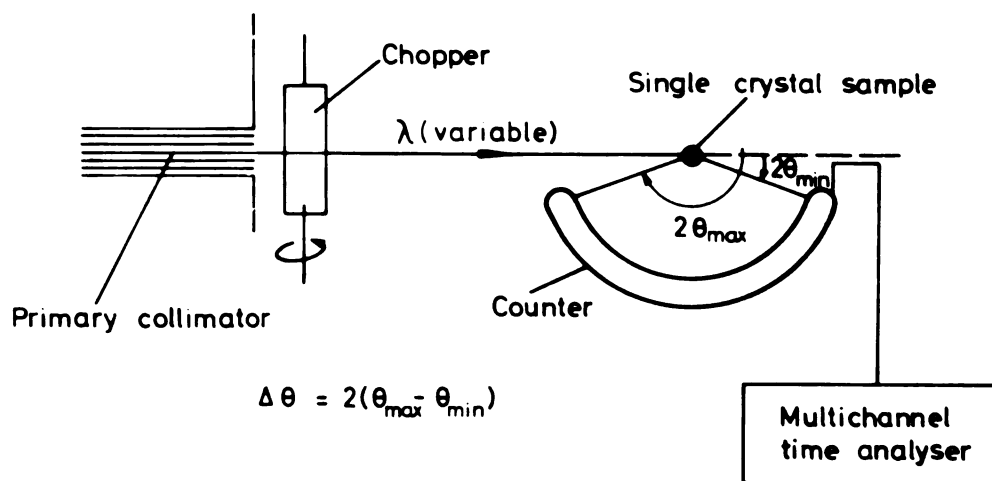


Fig. 8 The principle of the arrangement for TOF measurements of single crystals (32)

For the large $\Delta\theta$ (see Fig. 8) the experimental checks of the method described above were performed both at the DR 3 reactor at Risø and the EWA reactor at Świerk using a chopped beam of neutrons (32). Figure 9 shows a typical diffraction pattern obtained at Świerk for an aluminum single crystal.

For a small $\Delta\theta$ one obtains reflections from one plane and its higher orders. The experimental checks for this case have been made at Risø (32), Świerk and at the IBR reactor at Dubna. Figure 10 presents the results obtained in Risø for chromium-rhenium single crystals showing reflections from the magnetic "satellites" using the arrangement shown in Figs. 11-13. (33).

2.2. Comparison of the conventional and the TOF methods

2.2.1. Spectral distribution of incident neutrons

In order to measure structure factors by the TOF method it is necessary to know the spectral distribution of the incident neutrons. This could be done either

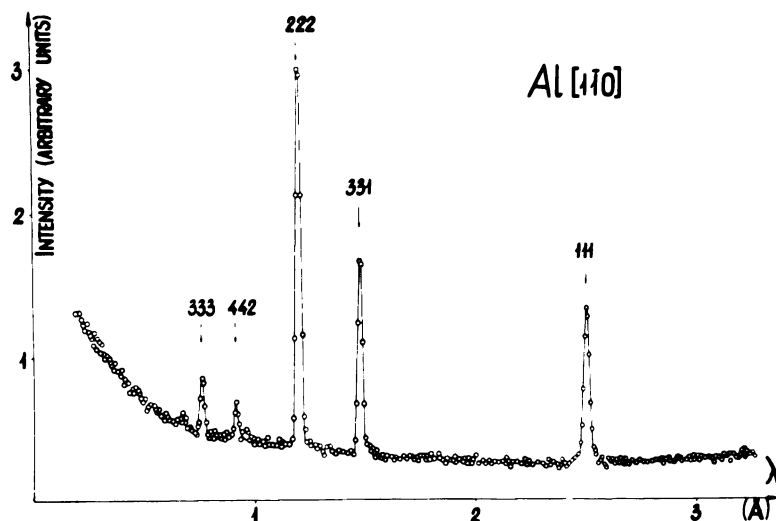


Fig. 9 Diffraction pattern of an aluminum single crystal obtained at the EWA reactor at Swierk

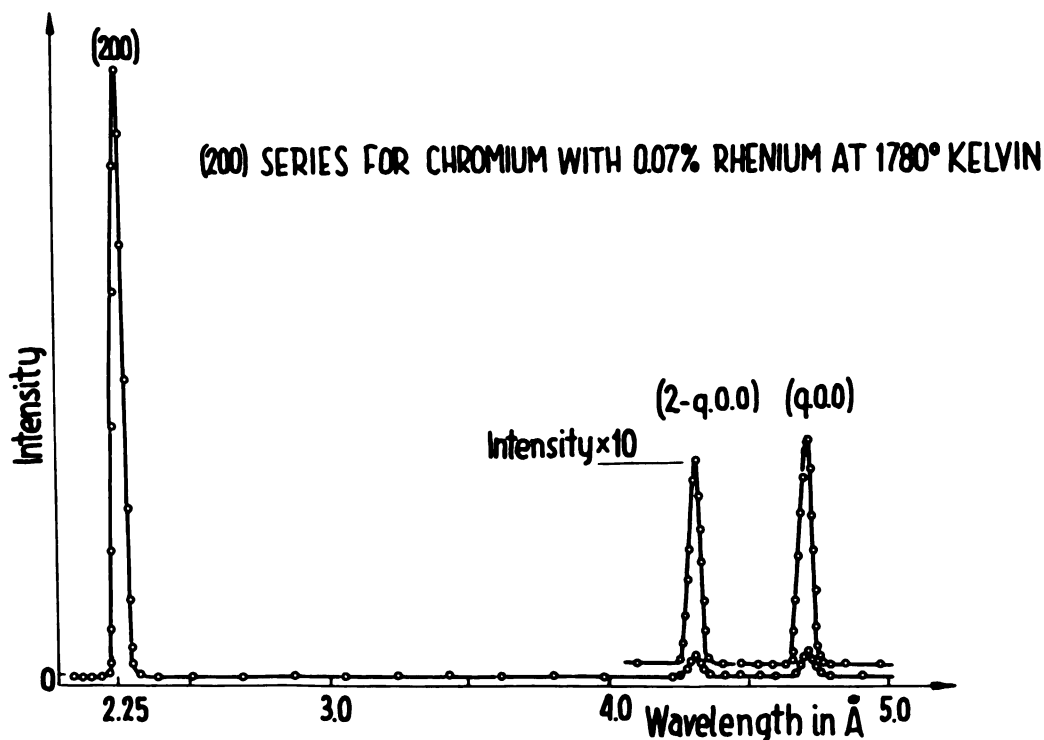
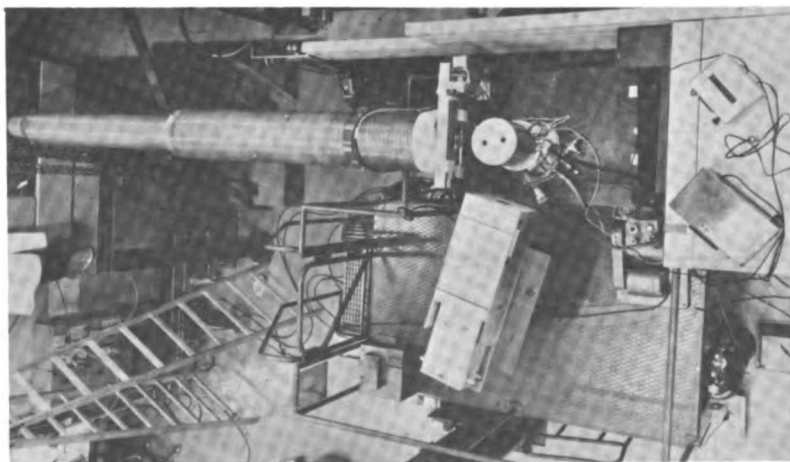


Fig. 10 Diffraction pattern of an chromium - rhenium single crystal obtained at the DR 3 reactor at Risø (33)



DR 3 Reactor (Risg) TOF Experimental Arrangement for Single Crystal Work

Fig. 11 General view of the TOF arrangement

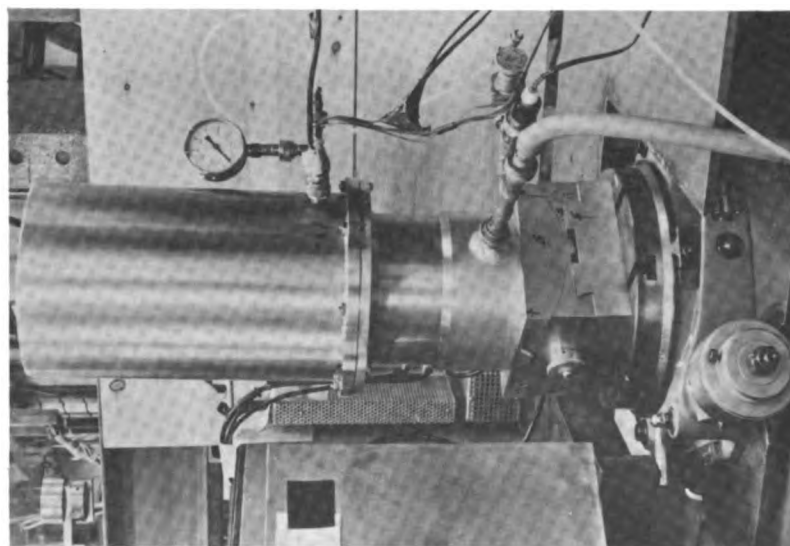


Fig. 12 The goniometer and cryostat

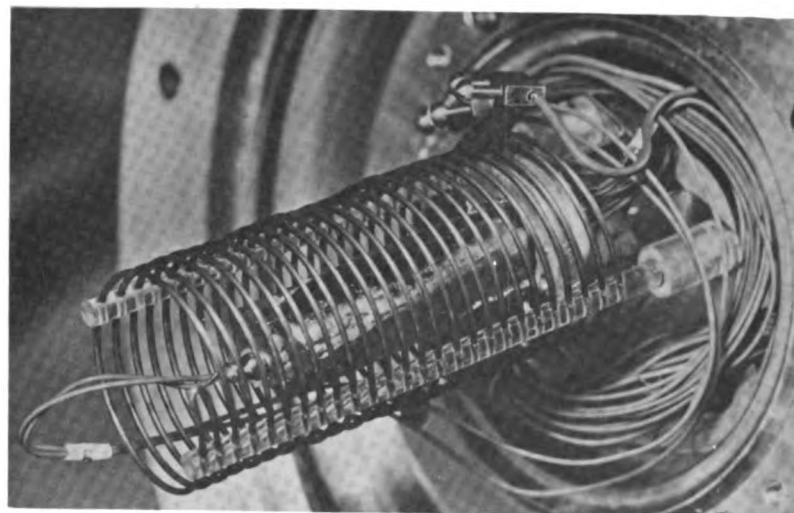


Fig. 13 The sample and heater

by using a monitor arranged in such a way that the spectrum of the monitor beam is observed at the same time as the peak in the scattered beam, or by measuring the spectrum independently. In the latter case this could be done either by measuring the spectral distribution of the direct beam by means of the same counter used for the scattered beam (no efficiency correction needed) or by calculating the spectrum from diffraction patterns of crystals with known structure. Figure 14 represents the agreement for the above two methods as used at the IBR reactor (21). Silicon powder was used as a calibrating substance.

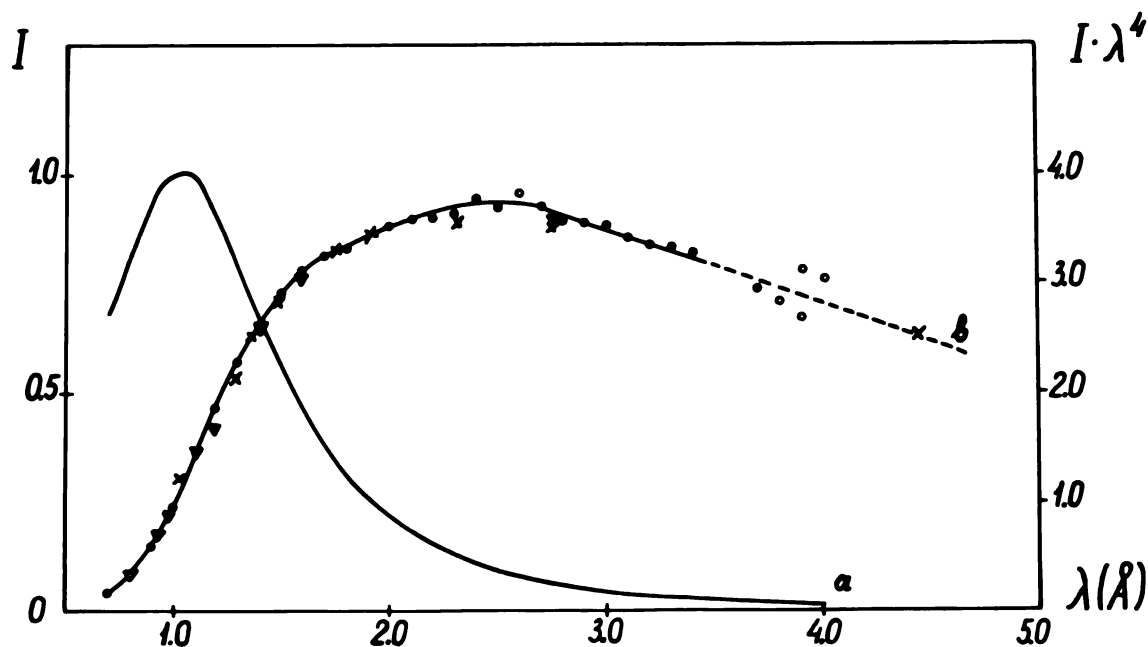


Fig. 14 The spectrum of the incident neutrons from the IBR reactor (curve a), the "effective" spectrum $I \cdot \lambda^4$ as calculated from curve a (curve b). The points were obtained by calculating the "effective" spectrum from diffraction patterns of silicon (21)

The need to know the spectral distribution in the TOF method is a disadvantage and an additional source of errors; however, the measurements performed until now have shown that this is not a serious obstacle.

2.2.2. Resolution and intensity

The resolution in the TOF method has been discussed by several authors (17,28,29,35-40). Figure 15, taken from Schwarz's (28) paper with some additional data from the IBR reactor, shows how the resolution in the conventional and TOF methods depends on interplanar spacing. It is easy to see that for large interplanar spacings the TOF method is preferable. The very good resolution obtained using the IBR reactor may be noticed. Figure 16 shows a similar comparison between a chopper

with different speeds and the IBR reactor (26). Notice that poisoning (1% H_3BO_3) of the moderator leads to a better resolution due to the shortening of the pulse (21). With a flight path of 50 m, now under construction at Dubna, it is hoped to obtain a resolution of about 0.5% for 4 Å interplanar spacing.

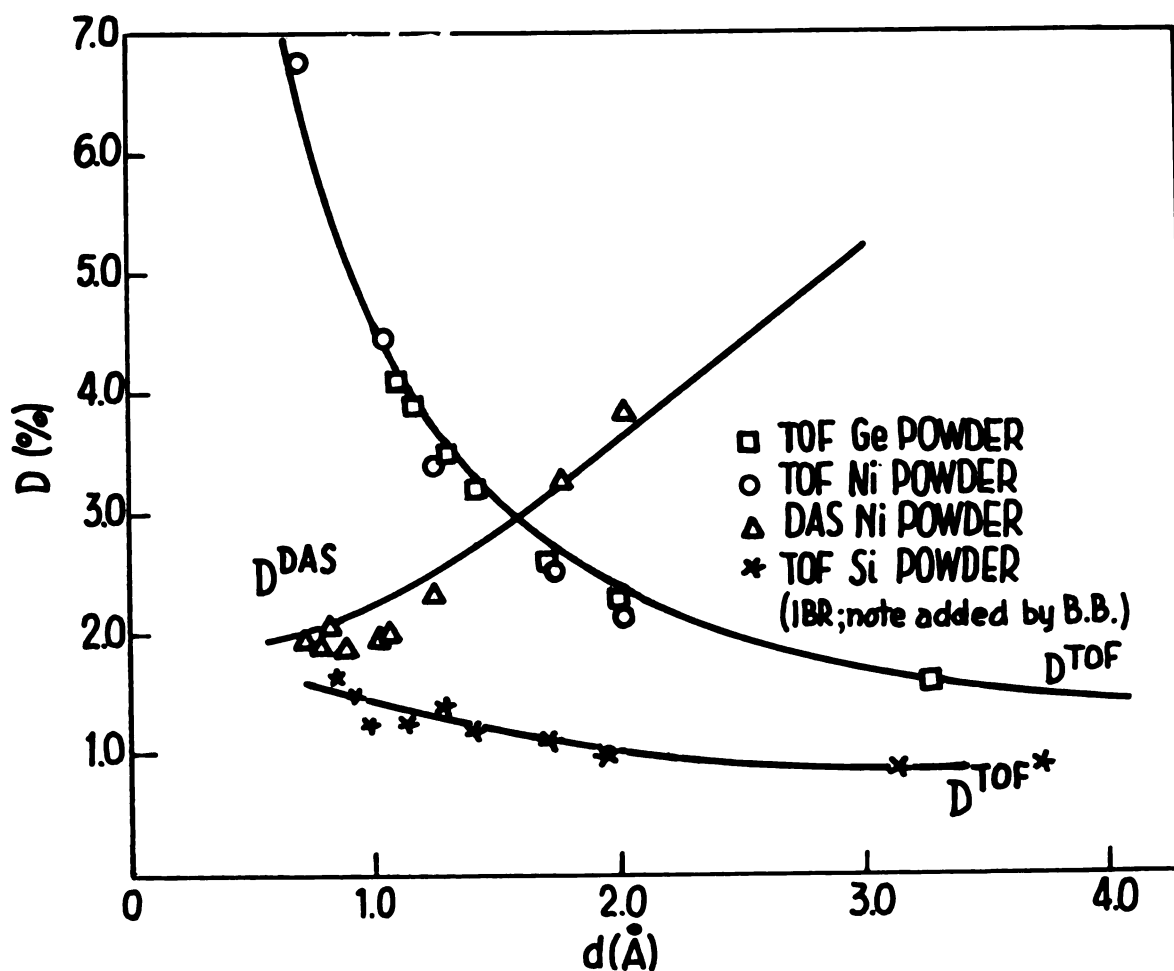


Fig. 15 Theoretical (solid lines) and experimental resolving power D of the TOF and double-axis spectrometer (DAS) for Ge and Ni as function of interplanar spacing taken from Schwarz's (28) paper. The experimental points for Si and the solid line drawn through them have been obtained at the IBR reactor

RESOLUTION AS FUNCTION OF WAVELENGTH

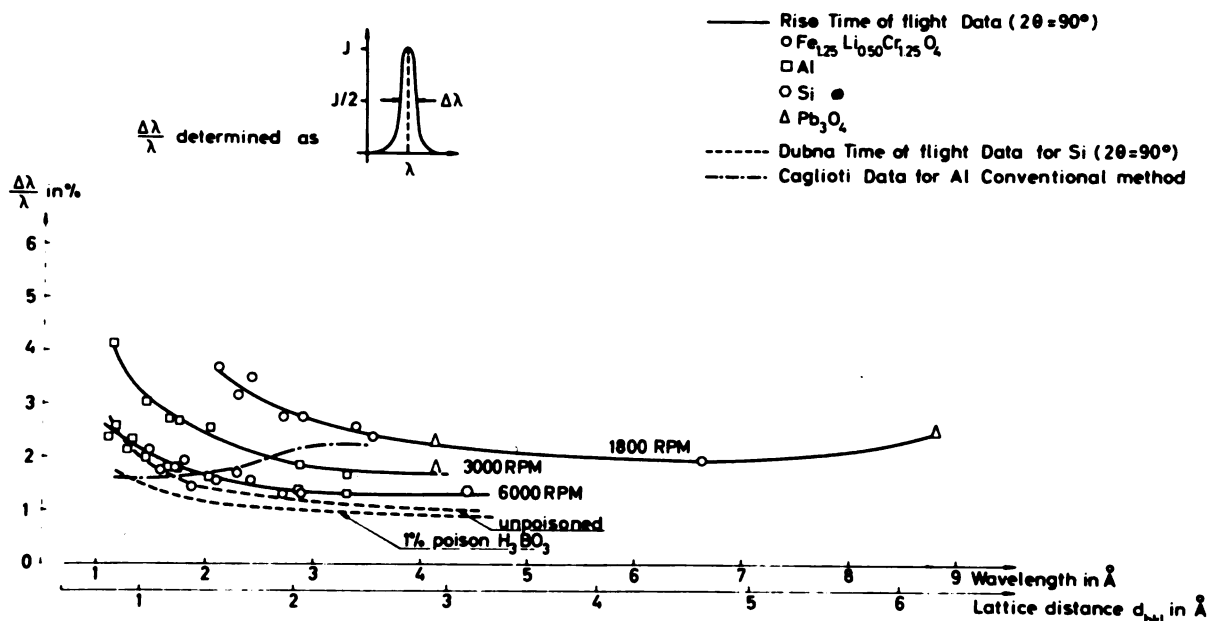


Fig. 16 Resolution as function of wavelength and interplanar spacing (26)

It needs not be stressed that, for many structure problems, a very high resolution is essential. However, almost every effort to improve the resolution leads to a decrease in intensity. In this connection Buras and Holas tried to arrange the experiment in order to obtain maximum intensity for a given, fixed resolution (37, 40).

The experimental arrangement can be described by a set of parameters. Figure 17 shows examples of these parameters: the distance L_r from the center of the source to the center of the sample, the dimensions of the source x_r and z_r and so on. Using these parameters, one can formulate mathematically the problem of how to choose the experimental parameters in order to get maximum intensity for a given, fixed resolution.

In the course of solving this problem, one takes advantage of the fact that the resolution depends in different ways on the various parameters. The same applies to the intensity. In this situation by a proper change of the parameters it is possible in general to increase the intensity without changing the resolution and finally obtain maximum intensity for a given resolution. Nowadays such a problem can be carried out using electronic computers, however, because of the large number of parameters this is time consuming.

The number of parameters for numerical calculations can be reduced by

dividing them into two classes: p and q . A parameter belongs to class q , if the integral intensity E is proportional to it, and if it appears in the resolution D as a term in q^2 . All other parameters belong to class p . Using this definition one can write:

$$E = f(p) \cdot \prod_{i=1}^n q_i \quad (1)$$

$$D^2 = \varphi(p) + \sum_{i=1}^n w_i(p) q_i^2 = \varphi(p) + \sum_{i=1}^n D_i^2 = C = \text{const.} \quad (2)$$

It can be shown for example that in accordance with this definition for a certain experiment (36)

$$\Delta X_r, \Delta X_s, \Delta X_c, \Delta Y_c$$

(Fig. 17) belong to class g , and

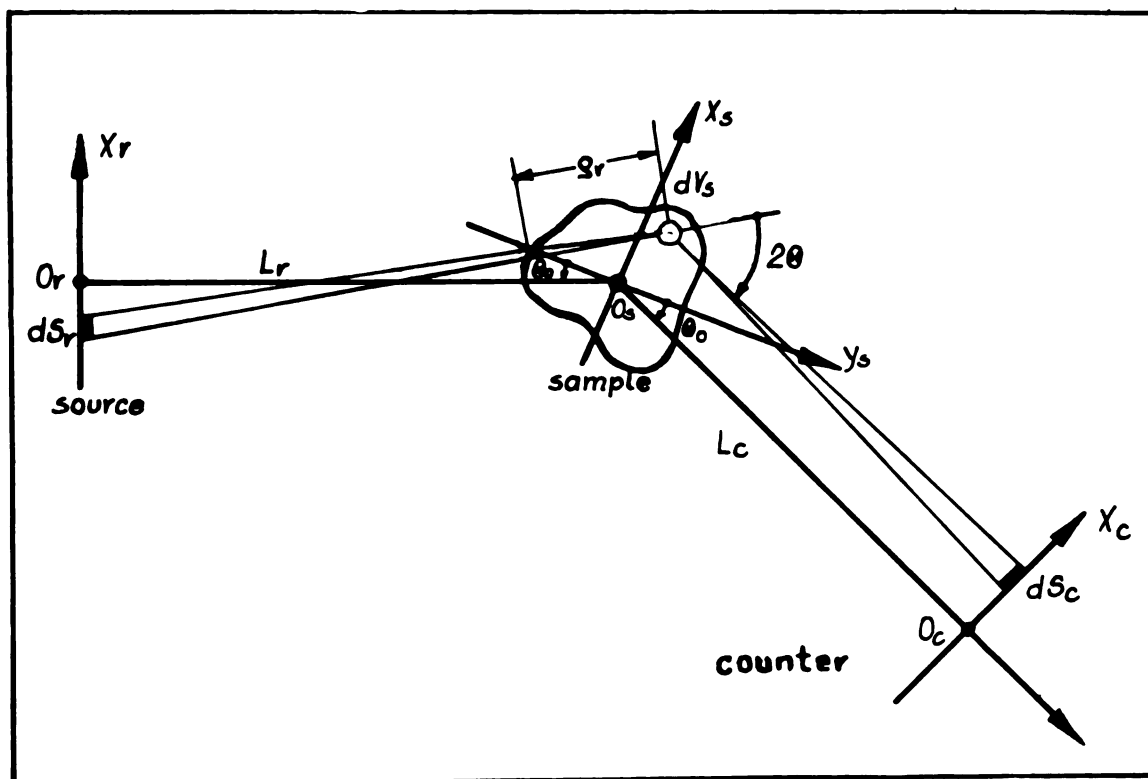


Fig. 17 Example of parameters describing an experimental arrangement for the TOF method (37)

$$\theta_0, L_r, L_c, \Delta Z_r, \Delta Z_s, \Delta Z_c$$

belong to class p. Returning to Eqs. (1) and (2) it is easy to show that optimization in respect to q class parameters can be done analytically. It can be proved (for the case when there are no limitations on the parameters q) that E is a maximum when

$$w_1(p)q_1^2 = D_1^2 = \frac{C - \varphi(p)}{n} \quad (\text{for all } i).$$

Therefore a computer is needed only to deal with the p-parameters which greatly reduces the time--and cost--for these calculations.

The procedure outlined above can be applied to any kind of experimental arrangement and enables one to find optimum conditions for any desired measurement. Analysis of previous experiments has shown that none have been done under optimum conditions. This demonstrates that in the TOF method there is still room for further improvements.

One such improvement, proposed by Holas (39), will be discussed in some detail.

Figure 18 shows that neutrons coming from different points on an extended neutron source and scattered by a point sample to a point detector correspond to different Bragg angles. A different Bragg angle implies a different wavelength and velocity. The neutrons coming from point A will enter the counter earlier than those from point O_r . This causes a broadening of the peak. In order to avoid this a small

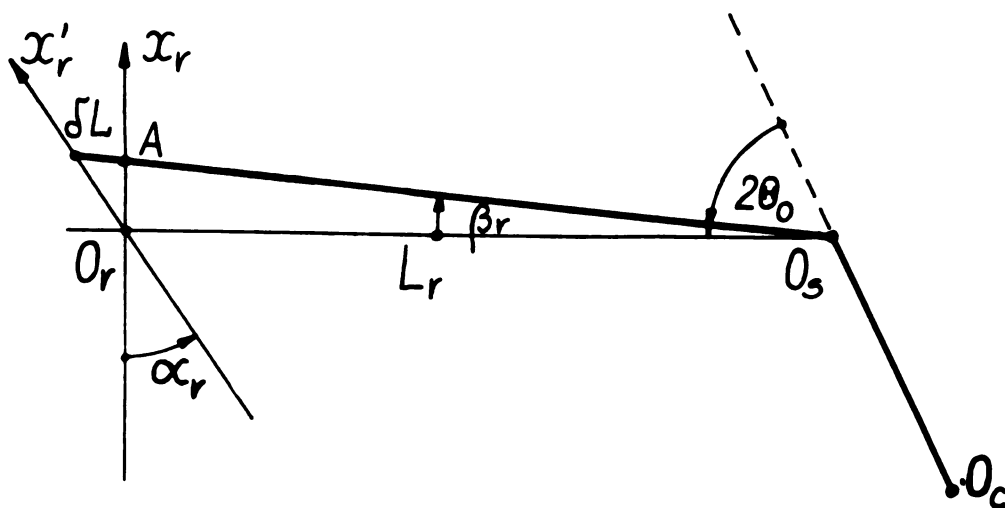


Fig. 18 Illustration of the focusing effect for a flat source: O_r - center of the source, O_s - point sample and O_c - point detector (39)

source or a collimator may be used, but this will cause a decrease in intensity. Another way to solve the problem is to incline the source, as shown in Fig. 18. This will make the flight path from point A longer, and for a proper angle α_r the "faster" neutrons will enter the counter at the same time as the "slower" ones in the central beam. It is easy to show that for every other point of the flat source this condition is satisfied for the same angle α_r . This effect--called focusing--enables the contribution to the resolution arising from the dimensions of the source to be neglected and one can safely increase the intensity by increasing the surface area of the source.

The same may be done with a flat counter. It can be shown that with a flat sample properly inclined the contribution to the resolution arising from the dimensions of the sample can be made negligible (39).

A preliminary experimental check made at the IBR reactor has shown that using focusing the intensity can be increased at least by an order of magnitude without loss in resolution. It should be stressed that by applying the focusing principle and by using scintillation counters that contributions to the resolution may be limited almost to the pulse time width of the source.

The comparison between the conventional and TOF method with respect to intensity and resolution was performed at Świerk. At the EWA reactor two neutron diffraction patterns with the same resolution and the same statistical error have been obtained using the conventional method and the TOF method. The time of exposure was a factor of about 5 less for the TOF method although this was not done in the optimum conditions, did not use the focusing principle and counts were collected from about 1/6 of the Debye-Scherrer ring only. This shows that the TOF method even with stationary reactors leads to shorter time exposures than the conventional one. To quote another figure--the diffraction pattern for silicon shown before (Fig. 5) with a resolution of about 1% for 4 Å can be obtained in 2.5 hours with the IBR reactor running at 5-kW average power. Taking into account optimization, focusing etc. one can calculate that with a similar reactor working on an average power of several MW one pulse will be sufficient in order to obtain a neutron diffraction pattern. This may enable "moving pictures" of structural changes to be taken.

2.2.3. Absorption and extinction

In the TOF method wavelength dependent absorption appears. In paper (21) it was shown for powders that corrections for absorption can be calculated easily. The extinction problem for single crystals is a serious one and not yet solved completely. Dietrich (41) elaborated at Risø a Monte Carlo method for calculation of extinction. This method has been used (with some slight modifications) at Świerk and checked experimentally. A good agreement between calculated and

measured extinction for aluminum single crystals was obtained. Work along this line is underway in Risø and Świerk.

2.2.4. Conclusions concerning conventional applications of the TOF method

The above discussions shows that the TOF method seems to be especially useful for the study of structures with large unit cells. The resolution is very high for large interplanar spacings which--for magnetic structure analysis--is the region where the magnetic structure factors are large. This makes the TOF method especially suitable for magnetic structure analysis, that has been demonstrated already by the work of Sosnowska et al. (23,24) on BiFeO_3 and on chromium-rhenium alloys (33) by Lebech and Mikke. The TOF is also advantageous for the study of small lattice distortions leading to peak splittings.

2.3. Special features of the TOF method

2.3.1. Constant geometry

The constant geometry obviously enables one to collect scattered neutrons from the whole Debye-Scherrer ring, which shortens the exposure time. In addition--and this is a more specific feature of the TOF method--the constant geometry may be useful in those cases where it is difficult to construct a sample holder which allows the scattering angle to be changed continuously. This may be the case when intense external fields (e.g., magnetic, high pressure) must be applied to the sample. The study performed by Brugger et al. (29) on samples under pressure up to 15 kbars using the TOF method (in order to obtain a diffraction pattern for KNO_3 phase IV and to observe the magnetic reorientation in $\alpha\text{-Fe}_2\text{O}_3$ with pressure) is an excellent example demonstrating this special feature of the TOF method.

2.3.2 Second and higher order contamination

The TOF method is free of second and higher order contamination, which makes it especially suitable for some applications as for example the study of "satellites" in chromium alloys (33).

2.3.3. Periodicity of the measurements

The periodicity of the measurements can be useful in cases where the influence of high intensity fields (not attainable in a steady state, but possible to obtain as repetitively pulsed fields, e.g., magnetic) on the structure is to be studied. By a coincidence in time of the neutron scattering events and the external field pulse, such a study is possible. This possibility has been mentioned (16,17) but no work has yet been reported.

By shifting the neutron pulse with respect to the external field pulse, the study of transient phenomena should be possible. Phenomena with relation times greater than 100 μsec would be in the observable range; however, using special means this could probably be made smaller.

The exposure times mentioned in section 2.2.2 would enable one to study very slow transient phenomena (e.g., phase transitions) in the range of minutes and hours.

These possibilities, again, have been mentioned (16,17) but no work has been reported yet.

3. LATTICE DYNAMICS STUDIES

As mentioned in the introduction the TOF method has been widely used for lattice dynamics studies (1,3-8). Here, only the so-called "inverted geometry" method for dispersion relation measurement will be discussed briefly. In this arrangement (Fig. 19) the direction of \vec{k}_0 is fixed, but its value is a function of time. The orientation of crystal and the direction of \vec{k} can be changed by rotating the crystal and the arm of the spectrometer. The value of \vec{k} is determined by the Bragg angle θ_B . This enables for one fixed position of the spectrometer the simultaneous measurement of the phonons for the same preset \vec{k} value. This method is especially suitable for measurements of dispersion relations for phonons propagating in a preset direction of the crystal under study. A preliminary check of this type of arrangement has been made at the IBR reactor and the results for resolution and intensity are promising. The advantages of using a pulsed source are similar to those mentioned in section 2.3.2.

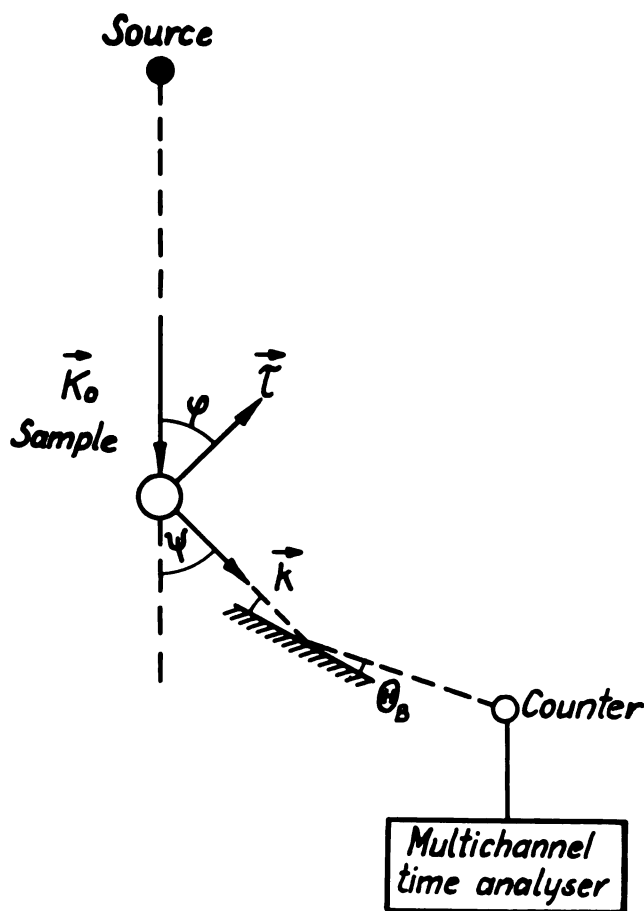


Fig. 19 Principle of the "inverted geometry" for dispersion relation measurements using single crystals and the TOF method

4. SIMULTANEOUS STUDIES OF STRUCTURE AND LATTICE DYNAMICS

As shown by Egelstaff and Cocking (42) for the case of graphite, the TOF method can be applied to the study of frequency distributions of crystals composed of coherent scatterers. The application of the TOF method both to structure and lattice dynamics studies offers a new and promising possibility of the simultaneous

structure and lattice dynamics study of solids (34). Figure 20 presents a proposal for an experimental arrangement for this purpose (in the case of powdered crystals). A pulsed collimated polychromatic beam of neutrons from the source Z is scattered by the powdered crystal sample S. The neutrons elastically scattered to the left are collimated and recorded by the counter L_1 connected to a multichannel time analyser A_1 . As it is known from the TOF method for powdered crystal structure analysis the information accumulated by the multichannel time analyser (providing that the spectral distribution of the incident neutrons is known) is sufficient for a structure determination. The neutrons inelastically scattered into the same direction contribute to the background only.

The neutrons inelastically scattered to the right will pass the beryllium filter only if their energy is smaller than 5 MeV (beryllium edge). As in the method described by A. Bajorek et al. (5,6), the information accumulated by the multichannel time analyser A_2 , connected to the counter L_2 , is sufficient to obtain--after using for example the procedure developed by Egelstaff and Cocking (41)--the frequency distribution.

In this way, using one sample, simultaneous measurements of structure and lattice dynamics should be possible. This idea, if feasible, can be extended - at least in principle - to single crystal studies. Figure 21 shows a possible arrangement for this purpose. In this case only several phonons will be measured.

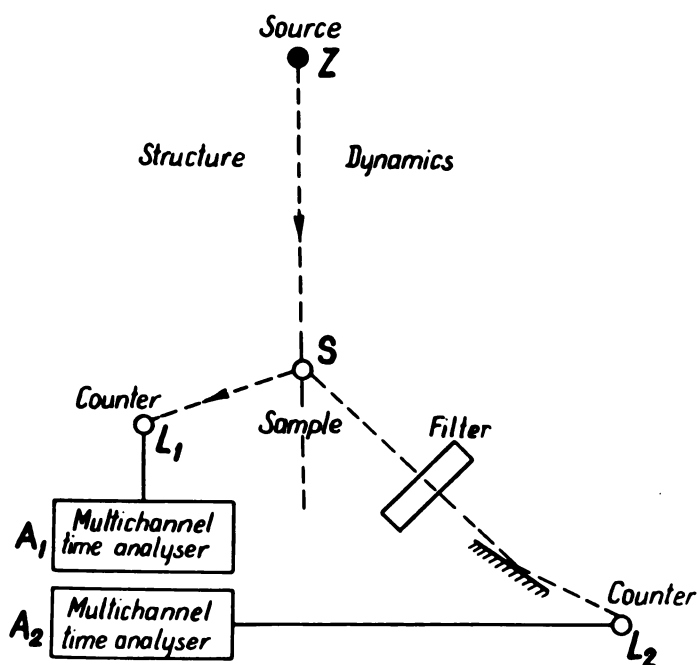


Fig. 20 Principle of simultaneous measurements of structure and lattice dynamics (frequency distribution) of powdered samples using the TOF method (34)

The proposed simultaneous measurements may be useful in studies of the relation between the structure of a solid and its lattice dynamics, especially in studies of phase transitions and the influence of external fields (e.g., electrical, magnetic, pressure) both on structure and lattice dynamics. The use of pulsed fields may be of special interest in this respect.

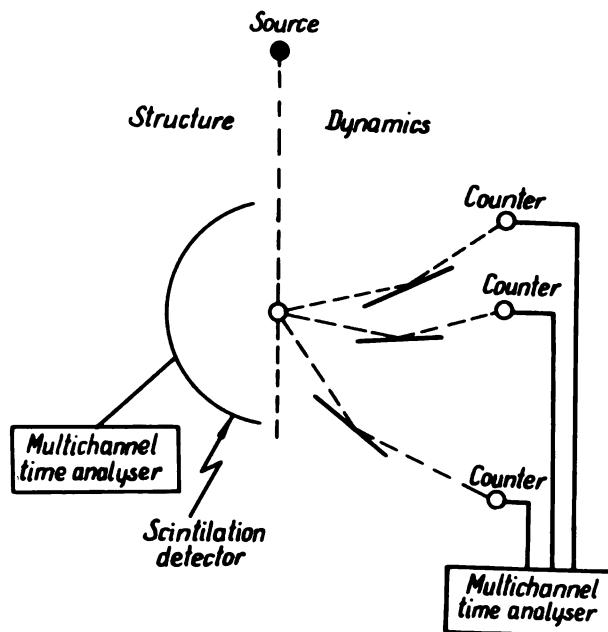


Fig. 21 Principle of simultaneous measurements of structure and lattice dynamics (several phonons) of single crystals using the TOF method

5. CONCLUSION

This brief review shows that the applications of the TOF method for the study of structure and lattice dynamics of solids open many new and unique possibilities. The essential condition for many of the discussed possibilities is the availability of a high intensity pulsed neutron source with a small pulse width and a neutron spectrum shifted towards long wavelengths. Among other things this will enable the use of small samples with high absorption cross sections.

Taking into account that some of the applications of pulsed neutron sources to the study of structure and lattice dynamics have a very short history (for structure analysis only 3 years) and the most intense pulsed neutron source available the IBR reactor--has only been working since 1960, one sees that we are now in a very interesting period when new possibilities are being opened and there is much hard work to be done in order to transform these ideas into reality.

6. ACKNOWLEDGEMENTS

The help of all colleagues, who have made available materials and information prior to publication, is greatly acknowledged.

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DISCUSSION
OF
CONTRIBUTED PAPER BY BURAS

MAIER-LIEBNITZ: In diffraction from single crystals, high resolution in neutron velocity is not basically necessary if good angular resolution is available. Therefore, a continuous beam with very moderate energy selection as proposed by Wilkinson or a moderated pulsed reactor like Whittmore's pulsed TRIGA reactor, together with a high angular resolution and 4π detection geometry may offer advantages in total intensity assuming equal peak fluxes in each case.

BURAS: We have discussed this subject before and I think that this is possible in principle; of course, one should prove it because the experiment is the final test of an idea. I would like to stress once more that there are two different approaches. One approach, as far as I understand it, is to measure angles with great accuracy; another approach is to measure wave lengths for a time interval. Most of you may know that in February 1965 at a meeting organized by the AEC, Shull from MIT gave a paper on the time of flight technique (CONF 650217). He made a very good point that you are simply trading mechanics against electronics.

DESIGN STUDIES FOR THE EXPERIMENTAL EQUIPMENT AT A VERY HIGH FLUX REACTOR

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SUMMARY: At research reactors, the intensity and the statistical accuracy can be improved by an increase of the reactor power as well as by careful optimalization of the experimental equipment. Both should be done during the development of a very high flux reactor. This paper includes four studies concerning the optimalization of various experiments: (i) The lay-out criteria for a bunch of neutron conductors originating at a cold source are discussed. A system with five separated neutron conductors is proposed in more detail. (ii) Two types of small angle scattering instruments are compared, namely a focusing and a non-focusing system. (iii) For certain classes of experiments, a high resolution double chopper and a triple axis spectrometer are compared with respect to intensity and applicability. (iv) A fission fragment separator of the focusing parabola type is proposed with a mass resolution of approximately 800. These investigations are theoretical studies. However, experimental experience obtained so far from work at the FRM and the FRJ-2 reactors was very important for many conclusions, mainly for the studies (i), (ii) and (iv). Most of the work has been stimulated by preliminary discussions of the beam hole program for the German-French very high flux reactor project.

1. A BUNCH OF NEUTRON-CONDUCTING TUBES FOR A COLD SOURCE

1.1 AIMS

Many experiments are best carried out with subthermal neutrons. As examples we mention inelastic scattering with high resolution (see Part.3), small angle scattering (see Part 2) and studies on n, γ reactions with low absorption cross section.

Thus a cold source should be utilized by many experiments at the

same time. To meet this need a "multipod" of conventional beam tubes was proposed some years ago [4]. We suggest the use of neutron conducting tubes (NT), which are of great advantage in the simultaneous utilization of a neutron source:

- a) Experiments can be set up along an interrupted NT ("tandem")
- b) The cross section of a NT can be shared by two NT's of different curvature ("fork")
- c) A bunch of several NT's directed on a common luminous source of limited cross-section can be housed in a single beam tube.

1.2 PROPERTIES OF NEUTRON CONDUCTING TUBES [2] [9]

The current of neutrons at the exit of a NT is given by the product of luminosity at the entrance and the transmitted solid angle Ω . For a straight, rectangular NT the latter is given by $\Omega = 4\gamma^2$ in sufficient approximation. The limiting angle for total reflection is

$$\gamma = \sqrt{\frac{Na}{\pi}} \cdot \lambda = \sqrt{4\pi Na} \frac{1}{k} = \frac{\Delta k}{k}$$

$$\Delta k_{Ni} = 1,07 \cdot 10^{-2} \text{ \AA}^{-1}$$

The luminosity be a Maxwell distribution [1]

$$dN(k) = \frac{\Phi_0}{2\pi} f \cdot F \frac{e^{-k^2/k_T^2}}{k_T^2} (2\Delta k)^2 k dk$$

$$k^2 k_T^2 = k_B \cdot T$$

$$\hbar k = m_n v_n$$

f stands for losses e.g. due to reflection.

F = cross section of NT.

An ideal NT would thus transmit constant intensity per ΔE excepting the Boltzmann-factor. As the NT must begin at some distance B from the cold source of limited area, this will no longer be true for small k.

The illumination ℓ of a NT can be calculated using a weighting function G(x,y) in the source plane. G is the convolution of a square R(x,y) of side length $2 B \gamma(k)$ in the source-plane, illuminating a plane element of the NT, and the cross section of the NT N(x,y). Representing the cold source by a luminosity function I we define the illumination as

$$\ell = \frac{\int G \cdot I \, d\mathbf{r}}{\int I \, d\mathbf{r}} = \ell_x \cdot \ell_y$$

As all functions are of the form $F(x,y) = f_x(x) \cdot f_y(y)$, all integrations can be simply performed. [3]. In the case of incomplete illumination one mainly loses neutrons of large angle of reflection implying a great number of reflections in the NT.

To reduce contamination of the cold neutron beam, direct sight for not totally reflected radiation can be prevented by curving the NT. The transmitted solid angle is reduced by 12% for these dimensions [2]:

$$\begin{aligned} \text{Radius of curvature } \rho &= 35\,000 \, a \, k^2 \\ \text{One "Length of sight" } L_I &= 530 \, a \, k \quad (k \text{ in } \text{\AA}^{-1}) \\ L_{II} &= 2 \cdot L_I \end{aligned}$$

a = breadth of NT.

1.3 GEOMETRY OF A BUNCH OF NEUTRON CONDUCTING TUBES

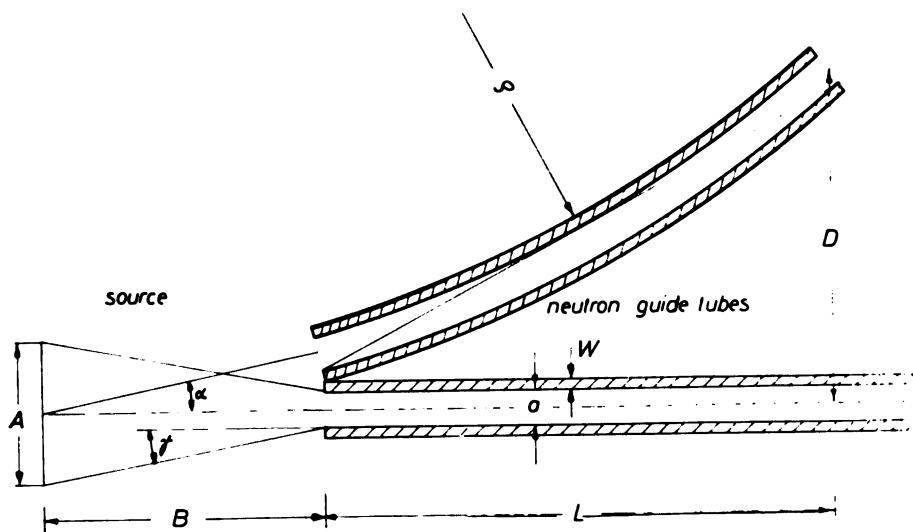


Fig. 1.1 Schematic drawing of two neighbouring NT's

Explanation of symbols in fig. 1.1:

- A. Linear dimension of cold source
- B Distance between cold source and NT
- a Breadth of NT
- α Initial angle between neighbouring NT's
- W Wall thickness of NT
- L Length of NT
- D Clearance between neighbouring NT's

The design of a bunch of NT's is more or less limited by 5 conditions:

1.31 Illumination in the (horizontal) plane of curvature

$$B \leq \frac{A-a}{2k} = (A-a) \frac{k}{2ak}$$

Technological reasons may call for larger B implying reduced illumination (see 1.2)

1.32 Neighbouring NT's do not penetrate. If all NT's beginn at the same distance from the cold source

$$\tan \alpha \geq (a+2u)/B$$

1.33 Taking curvature into account [2] the clearance to a neighbouring NT will be

$$D = L \left[\tan \alpha + 4L \left(\frac{a}{L_I^2} + \frac{a_N}{L_{IN}^2} \right) \right]$$

1.34 Unfortunately no simple relation for the length of the NT's as a function of k can be given if $L > L_I$. "Noise" from contamination of the totally reflected beam and environmental background both diminish with increasing length of NT's. This be compared to the "signal amplitude" given by the utilized fraction of transmitted intensity which is dependent on the type of experiment one has in mind:

Up scattering experiments imply an absolute breadth in energy ΔE .

For $k < k_T$ intensity is independent of k.

For small energy transfers ($\Gamma \sim k^2$) and elastic scattering relative resolution $\Delta E/E$ is decisive. Intensity $\sim k^2$.

In studies on n, γ reactions an integral over the neutron spectrum weighted by the probability for absorption gives intensity.

Double direct sight as quoted in 1.2 will probably be a criterion for maximum length. The average number of reflections is then 5.2. Losses due to imperfect surfaces will then amount to 17% with the 97% reflectivity now feasible.

1.35 An experiment should not be hampered by NT's on both sides.

This results in arrangements as given in Fig. 1.3

1.4 EXAMPLE OF LAYING OUT THE BUNCH. A bunch of five NT's is to be dimensioned for the wavelenths 3,5,7,15,25.Å ($k = 2.1, 4.25, 0.9, 0.42, 0.25$) The NT's are to have a common breadth of 5 cm.

For each k we chose an appropriate length. Considering background L_{II} was chosen for small k, while L_I was expected to suffice for large k (see Fig. 1,2)

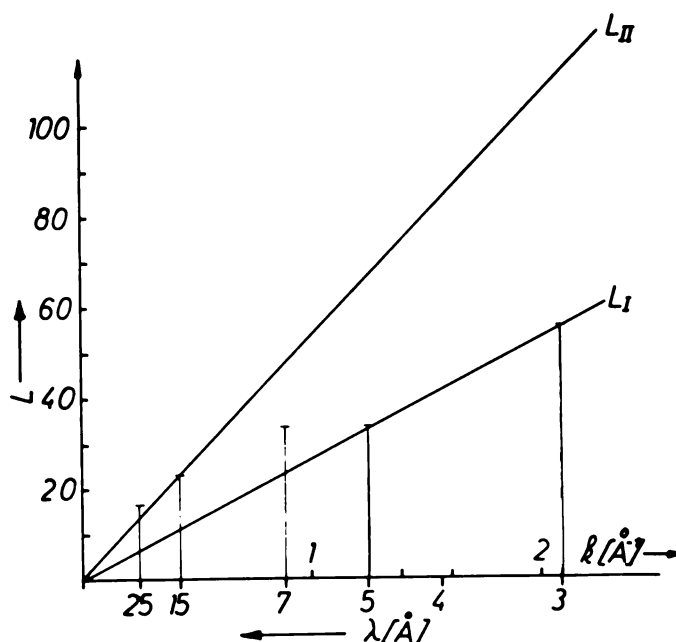


Fig. 1.2

In Fig. 1.2 L_I and L_{II} are represented as functions of k . Space for experiments will call for some distance between the exit ends of neighbouring NT's. (L in meters).

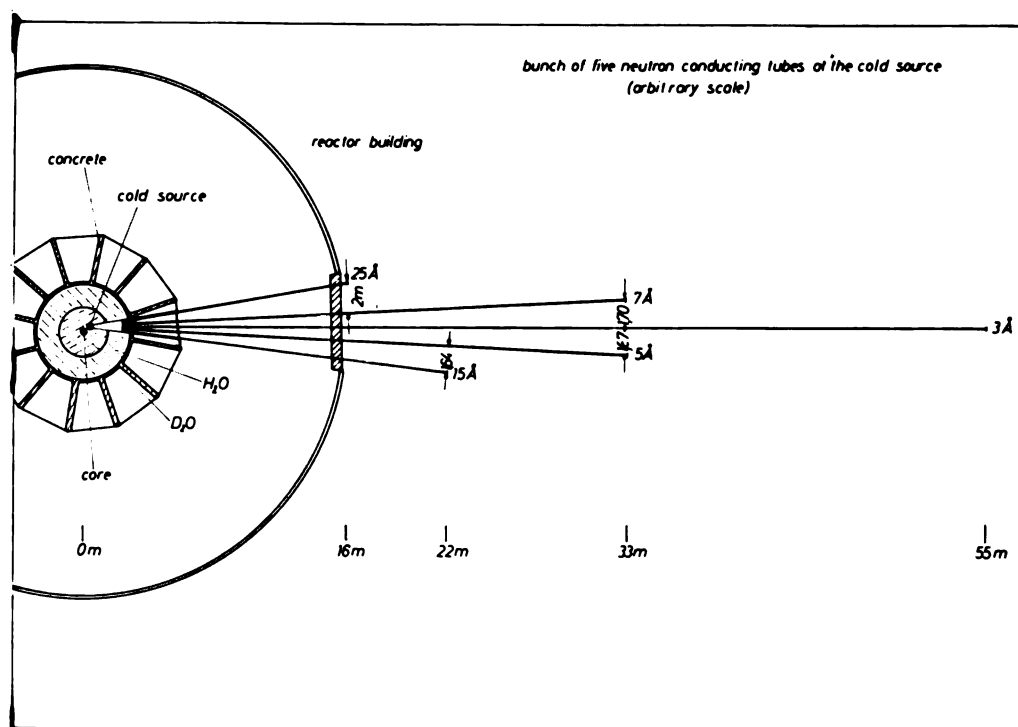
The cold D_2 source planned for the HFR Grenoble will have a diameter of 40 cm giving an available breadth for cold neutrons of approximately 20 cm.

The distance $B = 165$ cm was chosen to give total illumination as specified in 1.31 for the 25 Å beam. This seems to be reasonable in respect to the technological limitation given by the nuclear heating which will grow serious for material situated nearer the cold source. In order to facilitate an adaptation at a later date the NT's all begin at B .

The angle α between neighbouring NT's is constant and given by 1.32.

Resulting dimensions are given in Fig.13 and the following table.

Wave - length	Wave- number	radius of curvature	chosen length			clearance
	k		L_{II}	L_I	L	D
\AA	\AA^{-1}	km	m	m	m	m
25	0,25	0,11	12		16	2
15	0,41	0,3	22		22	1,6
7	0,89	1,4	47,5	23,7	33	1,7
5	1,25	2,7	66,5	33,2	33	1,6
3	2,1	7,7	110	55	55	



2. INSTRUMENTS FOR SMALL ANGLE SCATTERING EXPERIMENTS WITH VERY SLOW NEUTRONS.

2.1 APPLICATIONS. Small angle scattering experiments with slow neutrons have essential advantages as compared with x-rays. Because of the relatively low neutron absorption cross sections, rather long neutron wave lengths can be used ($\lambda = 5 \dots 20 \text{ \AA}$ and even higher). Therefore, it is easy to avoid double Bragg reflections which produce a strong background in the forward direction if crystalline samples are investigated. Furthermore, the experiments can be

extended to relatively small values of the momentum transfer:

$$K = \theta k_0 \approx 10^{-3} \text{\AA}^{-1} (k_0 = 2\pi/\lambda, \theta = \text{scattering angle})$$

or even smaller (see section 2.3). Finally, inelastic and magnetic scattering can be investigated with neutrons which is not possible with x-rays..

Many fields of applications seem to be of importance and need further improvement with regard to resolution or sensitivity; this is possible by using stronger neutron sources and by improving the instrumental equipment: (1) Critical scattering on ferromagnetic materials (e.g. [5]). (2) Critical scattering on gases which could be extended to smaller K -values than they have been obtained with x-rays so far. (3) Small angle scattering on dislocations in cold worked metals which gives information concerning the density and the arrangement of dislocations [6]. (4) Small angle scattering of neutrons on organic macro-molecules which is possible in spite of the strong proton incoherence. It might be interesting in a few special cases. (5) "Vortex scattering" in supra conductors [7] which needs a very high resolution. (6) Small angle scattering on atomic clusters in non-equilibrium liquid and solid solutions, and on large defect clusters (e.g. in irradiated quartz [8]). The intensity disadvantage of neutron sources as compared to x-ray generators can be overcome by a larger sample area, sample thickness, and quite generally, by the greater flexibility concerning the choice of the neutron wave length and the wave length resolution. In the following sections, we will discuss the relation between resolution and intensity for different instrumental systems, and we will say a few words on the practical experience which has been obtained so far.

2.2 COMPARISON OF DIFFERENT TYPES OF INSTRUMENTS.

The intensity as a function of resolution will be investigated for a simple slit system (fig. 2.1) and for a slit system with a focusing mirror (fig. 2.2). We assume that both systems include collimators with their slits perpendicular to the entrance slit to avoid corrections due to the finite slit height.

According to [1] we introduce the density of neutrons in phase space

$$n = (\phi/2\pi)(m/\hbar k_T^4) e^{-k_x^2/k_T^2} \quad \text{with } k_T = (2mk_B T)^{1/2}$$

(ϕ = neutron flux, T = moderator temperature); the number of neutrons crossing the sample with an area F per second is then:

$$\delta \mathcal{I}_0 = n k_0 F \delta k_x \delta k_y \delta k_z (\hbar^3/m) \quad (2.1)$$

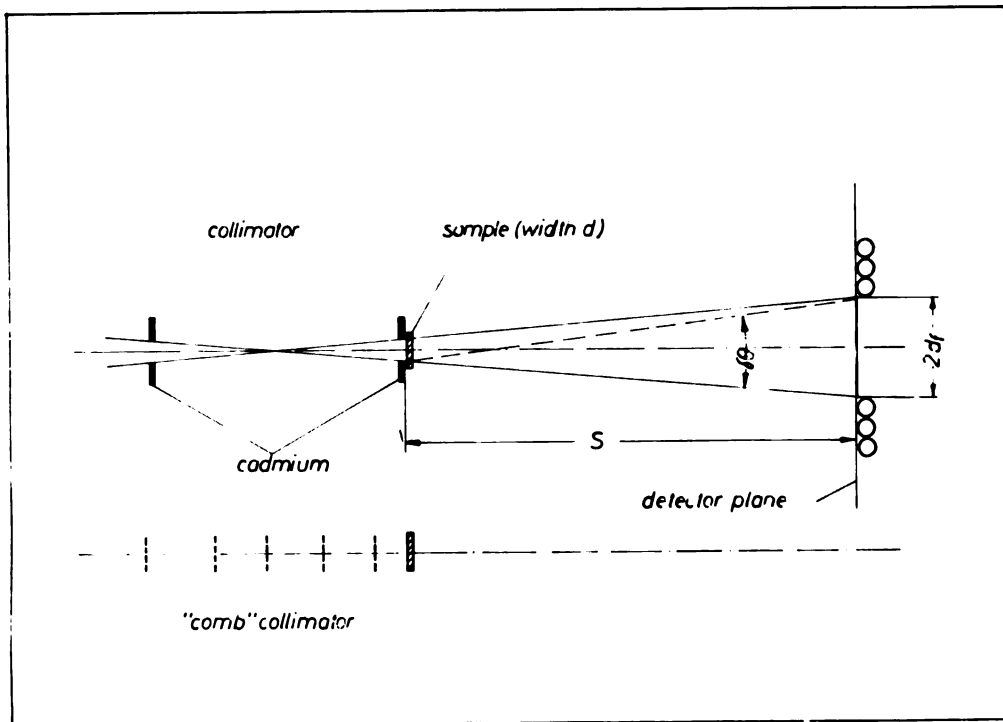


Fig. 2.1 Simple slit system; below: "Comb" collimator system.
 $\delta\theta$ is a measure for the resolution angle, $\delta\theta = \delta K / K_0$.

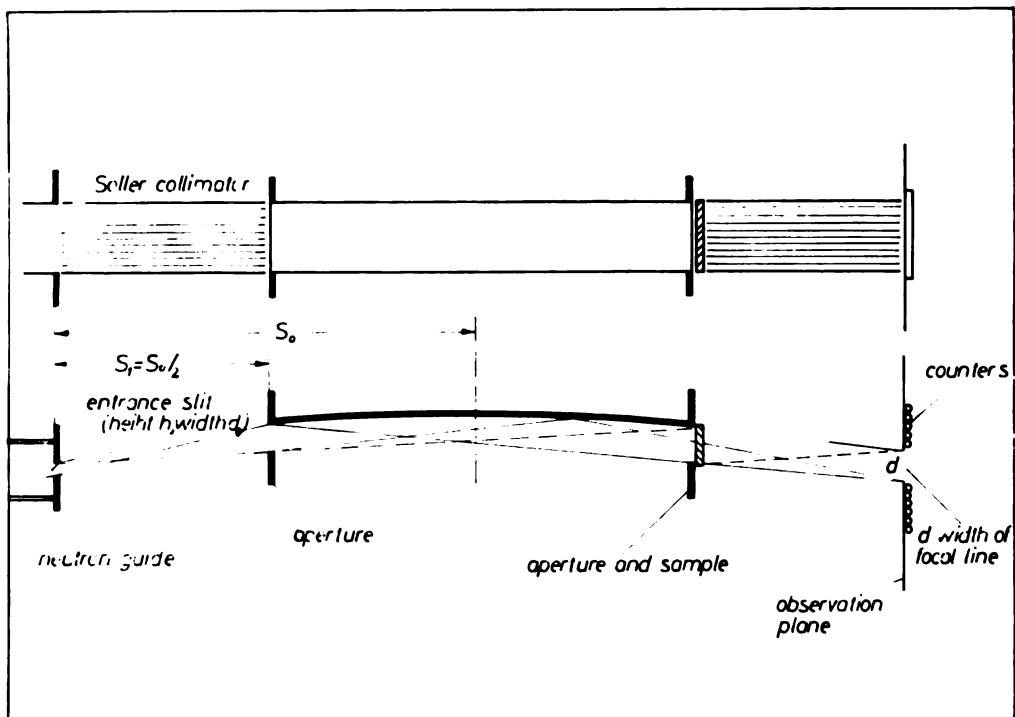


Fig. 2.2 Focusing system; above: Side view with horizontal Soller slits.

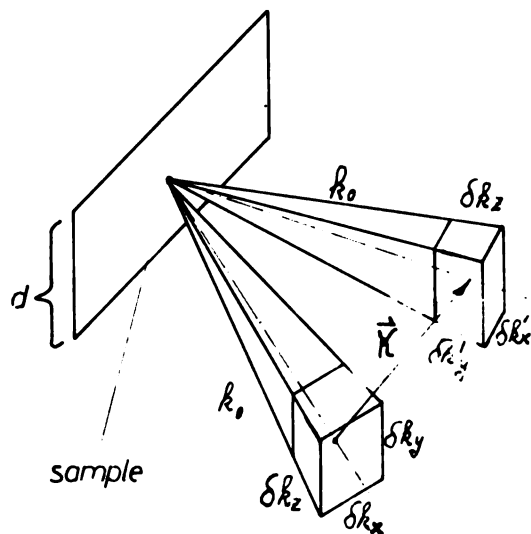


Fig. 2.3 Volume elements before and after scattering in phase space.

(figure 2.3) where $\delta k_x, \delta k_y$ and δk_z are the components of momentum spread of the neutrons penetrating the sample area, determined by the width of the focal line and the energy spread, respectively. The number of neutrons scattered into the detector is then

$$\delta \mathcal{F}_1 = \delta \mathcal{F}_0 W(\kappa) \delta \Omega = \delta \mathcal{F}_0 W(\kappa) \delta k'_x \delta k'_y / k_0^2$$

where $W(\kappa)$ is the scattering probability per solid angle.

The quantities after scattering are dashed.

To optimize the number of neutrons crossing the detector per second, all components of the momentum spread $\delta k_x, \delta k_y, \delta k_z, \delta k'_x, \dots$ contributing to the width of the momentum transfer, $\delta \kappa$, should be approximately equal. Therefore, we should apply for

$$\delta \kappa / \sqrt{i} \approx \delta k_x \approx \delta k_y \approx \delta k'_x \approx \delta k'_y = \theta \delta k_z \quad (2.2)$$

where $i = 5$ is the number of such independent contributions. We match the "longitudinal" contribution δk_z in the middle of the κ -range which has to be investigated (say at $\kappa = 5 \delta \kappa$). Therefore we have to put

$$\delta k/k_0 \approx 1/5\sqrt{2} = 0,09; \delta E/E_0 \approx 0,18.$$

It is easy to achieve such an energy resolution by a mechanical slit selector (in most x-ray experiments, the energy resolution is higher than necessary). Therefore, if the system is wholly optimized, the number of neutrons scattered into the detector is,

$$\delta \mathcal{F}_1 = \left(\frac{h^4}{m}\right) n(k_0) w(k) F \frac{\delta k^4}{25} \cdot 0,09 \quad (2.3)$$

This quantity is proportional to the 4th power of the momentum resolution and independent of k_0 excepted the Boltzmann factor in $n(k_0)$. Now we have to impose certain geometrical conditions to obtain the components of the momentum spread required by eq. (2.2).

For a simple slit system, the distance between sample and detector has to be

$$S = d k_0 \sqrt{2} / \delta k \quad (2.4)$$

We assume that the sample width, d , is given by experimental reasons, say, $d = 2$ cm, and chose a wave length of 20 \AA . Then, for a required resolution of $\delta k = 10^{-3} \text{ \AA}^{-1}$ we find
 $S = 14$ meters.

Including the collimator before the sample. the whole system is, therefore, about 30 meters long. The length of the system could be strongly reduced without changing the intensity considerably if multi-slit "comb" collimators are used (figure 2.1). However, if a collimator is placed between the sample and the detector plane, it is not possible to measure the intensity pattern simultaneously which is the inherent disadvantage of the short system.

For a focusing slit system (see [9]), the sample width d is connected with the critical angle of the mirror surface, γ_m , approximately by

$$S_1 \approx d / c \gamma_m = d k_0 / c \Delta k_{crit} \quad (2.5)$$

(S_1 = distance between sample ^{and detector,} and Δk_{crit} is a constant depending on the refractive index of the mirror material.) For glass, $\Delta k_{crit} = 0,0063 \text{ \AA}^{-1}$. $c \approx 0,9..1$ depending on the special geometry [19]. It is easy to "illuminate" this angle $c \gamma_m$ if the entrance slit is connected with the reactor by a glass or nickel neutron guide. Using, as above, $d = 2$ cm, $\lambda = 20 \text{ \AA}$, one finds.

$$S_1 \approx 1 \text{ meter}$$

This corresponds to a mirror about 2 meters long and a total length of the system of about 4 meters.

The width d_0 of the entrance slit has to be adapted to the required resolution by means of

$$d_0 \approx S_1 \delta k / k_0 \sqrt{V} = 0,14 \text{ cm for } \delta k = 10^{-3} \text{ \AA}^{-1}.$$

Image errors have been neglected (see [9]).

We conclude: We have assumed the same phase space density, sample area, and resolution, for both systems; the phase space elements are equal. Therefore, both have the same count rate at the detectors. The disadvantage of the non-focusing system is its great length, and also the corresponding great height of the detectors (excepting cases where the samples have to be very small due to experimental restrictions). The focusing system has inherent difficulties because of irregularities at the mirror surface which reduce the resolution, especially if the scattering probability is small (see section 2.3). If, in the focusing case, the sample size is not restricted, the area F could be increased further by increasing the neutron wave length. However, there is a restriction because of the increasing absorption in the sample and an increasing number of reflections in the neutron guide.

In cases where the scattering law depends also on the azimuth (e.g. for dislocation or defect scattering on single crystals), a system with circular apertures should be used. So far, no experimental experience exists on toroidal mirrors for circular focusing systems. For non focusing systems of great length, the influence of gravity on resolution becomes serious [19].

Instead ^{with} multi-detectors, the two-dimensional intensity pattern should be measured with a photo technique. Good success has already been obtained with scintillation layers containing Li^6 in contact with a photographic emulsion [10]. The development of one- and two-dimensional counters of large area would be of great importance.

2.3 EXPERIMENTAL EXPERIENCE.

Experience has been obtained with a non-focusing system at the FRM-reactor (Munich) [6]. Figure 2.4 shows the intensity distribution across the observation plane. At the FRJ-2 (Jülich) a mirror system

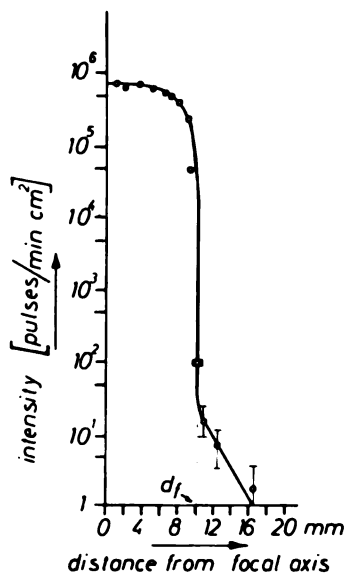


Fig. 2.4 Intensity distribution across the detector plane for a simple slit system. d_f gives the edge of the geometrical shadow (see figure 2.1).

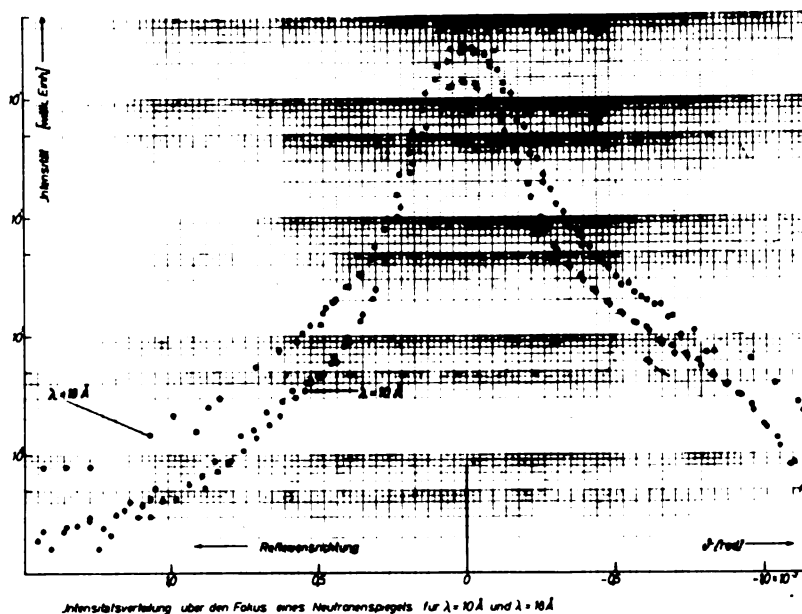


Fig. 2.5 Intensity distribution across the focal plane for a focusing mirror (mirror length 2 meters; mirror height 15 cm, focal length about 3 meters, entrance slit 3mm wide, 10 cm high). The pattern has been measured at different wavelengths λ . (E. Taglauer).

is under investigation. The mirror is 2 meters long and 15 cm high. It consists of 20 plane glass plates (quality of commercial mirror glass). The curvature (400 meters radius) has been adjusted by optical methods. The neutronic focus (figure 2.5) has wings which are stronger than those of figure 2.4 because of surface irregularities. The width of the halo depends only very weakly from the wave length. Therefore, we assume that the halo is essentially not due to diffraction and the irregularities are expected to be smooth and extended and an improvement can be expected by using better surfaces. Image errors due to the non-curved segments and due to the inclination of the focal plane induce a broadening not more than a factor of two.

3. COMPARISON OF A TRIPLE AXIS WITH A TIME-OF-FLIGHT SPECTROMETER FOR HIGH RESOLUTION EXPERIMENTS

3.1 GENERAL

Different instruments can only be compared for a certain class of experiments, and not in general. In the following, experiments are considered which have in common that the energy resolution should be high whereas the momentum resolution can be relatively poor. Back scattering should be used whenever possible to obtain a good momentum resolution in spite of large solid angles (see (17)). This means

$$\mathbf{k} = \mathbf{k}_0 + \mathbf{k}_1$$

where $\hbar\mathbf{k}$ is the momentum transfer and $\hbar\mathbf{k}_1$ is the momentum of the neutron after scattering, $\hbar\mathbf{k}_0$ the same before scattering.

(1) Quasi-elastic scattering in a wider sense: (i) Incoherent scattering on diffusing atoms in liquids shows a quasi-elastic line with a half-width of approximately

$$\Gamma = 2\hbar\Lambda\mathbf{k}^2 \quad (3.1)$$

as long as the momentum transfer $\hbar\mathbf{k}$ is sufficiently small. Normally, the diffusion constant Λ is of the order of 10^{-5} cm²/sec, and smaller in viscous substances. (ii) Under certain conditions a formula similar to (1) is valid for coherent scattering on critical fluctuations in gases, solids, and paramagnets at small \mathbf{k} values, and at non-critical fluctuations in paramagnets. In these cases the appropriate constant Λ is much greater (10^{-4} .. 10^{-3} cm²/sec for critical scattering).

Because the \mathbf{k} -dependence of the quasi elastic scattering is smooth

in all cases, the \mathbf{k} -resolution can be relatively poor, namely about

$$\delta k/k \approx \frac{1}{2} \hbar \delta \omega / \Gamma = 0.05 \dots 0.1$$

where $\hbar \delta \omega$ is the energy resolution and $\hbar \delta \omega / \Gamma = \beta$ is assumed to be 0,1..0,2. As will be seen later, backscattering is only possible for the case (i).

(2) Investigation of phonon lines in crystals for selected and fixed wave vectors in the reciprocal lattice, especially their width, their shift with temperature etc. An important case is shown in fig. 3.1 where

$$k_i + k_f = 2\pi\tau - q \quad (3.2)$$

In this situation, a relatively great solid angle is allowed without deteriorating the resolution too much ("focusing"). The experiment according to eq. (3.2) is restricted to longitudinal phonons. A 90° scattering experiment allows focusing and the observation of both polarizations.

No detailed discussion need to be performed on the incoherent inelastic scattering on lattice vibrations. Also in this case the \mathbf{k} -resolution can be poor because the \mathbf{k} dependence is necessary for multi-phonon correction only.

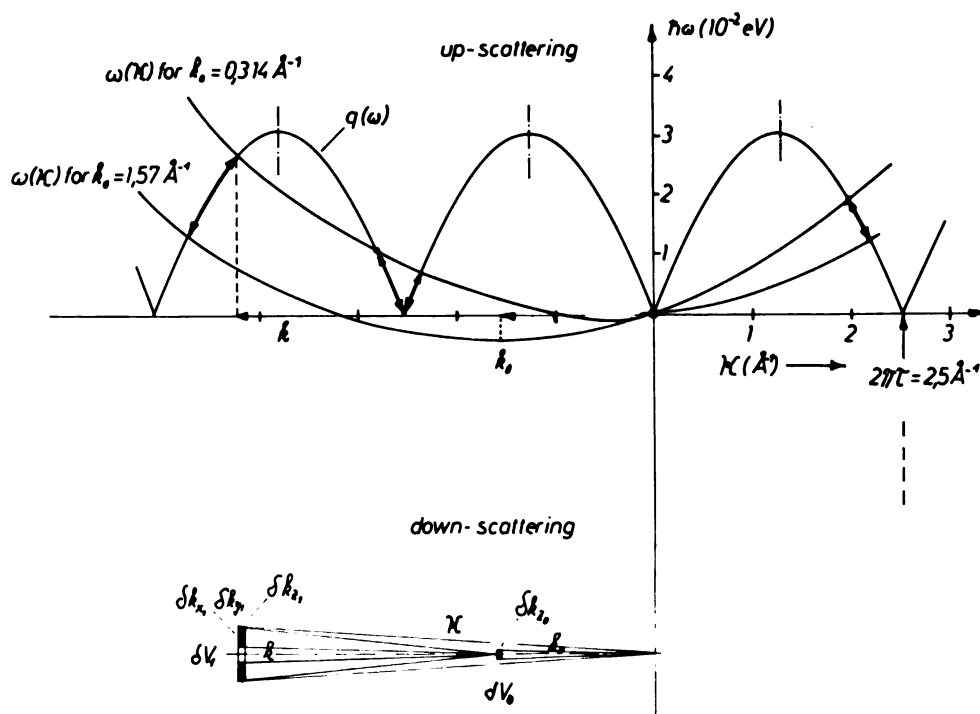


Fig. 3.1 Backward scattering experiment ($\theta \approx \pi$), showing the volume elements in phase space, δV_0 and δV_1 , before and after scattering,

respectively (below). The corresponding equation of momentum conservation for one-phonon scattering (eq. 3.2) is represented above. Thin lines: Relation between momentum transfer $\hbar K$ and energy transfer $\hbar\omega$ for two different wavelengths of the incoming neutron. A hypothetical dispersion law $q(\omega)$ along a symmetry direction with a maximum phonon energy of 0,031 eV and $\omega = 2,5 \text{ \AA}^{-1}$ has been assumed. Solid line indicates that portions of $q(\omega)$ which can be covered by incoming wave lengths between 4 and 20 \AA . The minima of the $\omega(K)$ parabola are situated at $\hbar\omega = E_0$. Foreward scattering ($\theta=0$) is also indicated (right side) which is not discussed in the text.

3.2 INSTRUMENTS UNDER CONSIDERATION

Subject of the study are a double chopper time-of-flight instrument and a triple axis spectrometer. Both instruments are assumed to be situated at the end of a bent neutron conducting tube originating at the cold source of the reactor.

For the choppers, shielding material can be avoided because the fast neutron background in the beam is very low (because of the great distance from the reactor). Large detectors are also permissible. In order to have a large solid angle between the choppers in spite of their relatively great distance, the flight path consists of a neutron conducting tube. A detailed analysis of this time-of-flight instrument has been given by Scherm [11]. In the case of the triple axis spectrometer, Bragg angles close to π are allowed because of the low background.

3.3 INTENSITY AND LAY-OUT

As can be shown [1] the scattered intensity of a neutron spectrometer is proportional to the total thermal flux and to the factor

$$F e^{-E_0/k_B T} \delta V_0 \delta V_1 f_i f_r \quad (3.3)$$

F is the sample or beam area, $k_B T$ is the (effective) temperature characteristical for the cold source spectrum, and f_i, f_r are loss factors which will be explained below, $\delta V_0, \delta V_1$ are the volume elements in phase space before and after scattering, respectively, where (see fig 3.1)

$$\delta V_0 = \delta k_{x0} \delta k_{y0} \delta k_{z0}, \quad \delta V_1 = \delta k_{x1} \delta k_{y1} \delta k_{z1}$$

$\delta k_{x0}, \delta k_{y0}, \delta k_{z0}$ are the widths of the momentum vectors before, those with subscript 1 after scattering. Because the instruments are situated at the end of a neutron conducting tube one has $\delta k_{x0} = \delta k_{y0} = \Delta k_{crit}$

($= 2,14 \cdot 10^{-2} \text{\AA}^{-1}$ for a Nickel neutron guide), independent of the neutron energy. δk_{20} and δk_{21} are determined by the required resolution widths δE_0 and δE , which will be introduced as a fraction β of the energy transfer $\hbar\omega$ which has to be investigated:

$$\delta k_{20} = (k_0/2)(\delta E_0/E_0) = k_0(\hbar\omega/E_0)(\beta/2)$$

If the energy after scattering, E_1 , is a factor a^2 greater than E_0 , one should aim at $\delta k_{21} \approx \delta k_{20}/a$

For quasi elastic scattering we have $a \approx 1$.

Comparing both instruments we make a few assumptions which are dependent on the technical aspects of the experimental set-up:

(1) In the case of the chopper, the beam area is essentially restricted by the pulse length and the available rim speed, and in the case of the triple axis instrument by the sample and crystal sizes. For the triple axis instrument we propose $F = 50 \text{ cm}^2$. For the chopper and for quasi elastic scattering experiments we use

$$F = 2,5 \times 10 = 25 \text{ cm}^2$$

For inelastic scattering experiments, E_1 might be considerably greater than E_0 and shorter pulses are needed. This requires a smaller slit width and we take $F = 1,25 \times 10 \text{ cm}^2$. With a slit speed of 500 meters/sec and a disk chopper, a burst half-width of 50 μsec (with 2,5 cm slit width) can be obtained, and of 12,5 μsec with two reverse choppers of 1,25 cm slit width.

(2) In our case, the solid angle after scattering is not seriously restricted by momentum resolution if back scattering can be achieved. For the time-of-flight instrument a certain restriction comes from costs and from background.

We use $\delta k_{x1} \delta k_{y1} = k_1^2 4 \cdot 10^{-2}$

this means e.g. a 4 square meter detector at the end of a 10 meter long flight path. Of course, this solid angle should be subdivided to enable 2nd order corrections due to the momentum spread. For the triple axis instrument we take a divergency after scattering of the order of the mosaic spread of a "poor" crystal (half width about 1°) and a vertical divergency of 6° , therefore

$$\delta k_{x1} \delta k_{y1} = k_1^2 \cdot 1,8 \cdot 10^{-3}$$

(3) Losses. For the triple axis instrument we put $f = 0,25$ due to reflectivity losses on both crystals. For the double chopper we can write $f \approx 1$.

3.4 THE CHOICE OF THE NEUTRON ENERGY E_0 , THE OBTAINABLE RESOLUTION, AND THE TIME-UTILIZATION

For the time-of-flight instrument and for quasielastic scattering, the energy before scattering, E_0 , should be as small as possible (i) to allow back scattering, or, at least, to allow great scattering angles so that the required momentum resolution δK can be obtained in spite of the assumed large solid angle after scattering, and (ii) to obtain a high time-utilization (see below). On the other hand, E_0 has to be sufficiently great to facilitate the evaluation of the profile of the scattering law, and one should have

$$E_0 \geq 2..4\Gamma \quad (3.4)$$

Unfortunately, this equation, together with eq. (3.1) shows that back scattering is possible only if $\Lambda < 2 \cdot 10^{-5} \text{ cm}^2/\text{sec}$. The whole situation is represented in fig. 3.2 for two typical examples. It should be noted that back scattering, and therefore the relatively large solid angle cited before, is not allowed in the case of fluctuation quasi-elastic scattering.

A certain restriction towards low energies is the increase of the reflectivity losses in the neutron guides. They increase with the number of reflections which is approximately proportional to the wave length. According to the present experience one should not go beyond, say, 20 or 30 Å for a 20 meter long guide tube (for more details see [2]).

The time utilization is given by the ratio

$$f_T = \tau' / \tau$$

where τ is the pulse distance and τ' is the time interval during which relevant information on the scattering law is collected. According to eq.(3.4), and allowing a certain amount of frame overlap in the wings of the quasi elastic line, one can assume

$$f_T \propto \Gamma / E_0$$

which is greatest for backward scattering ($f_T = 1/2 \dots 1/4$)

For the cited minimum pulse length of 50 μsec combined with flight

paths of 5 meters between the choppers and after the sample, it is easy to choose E_0 which fulfills the above requirements concerning the energy resolution for quasi elastic scattering (see fig. 3.2).

In the case of inelastic scattering experiments, a flight path between the choppers of 5 meters, of 10 meters after the sample, and a pulse length of $12,5 \mu\text{sec}$ would be desirable to obtain a resolution of 0,7 per cent (with $E_1 = 0,025 \text{ ev}$).

At the triple axis instrument, the wave length is restricted by the lattice distance of the crystals. At these relatively short wave lengths it might be difficult to obtain the quoted momentum resolution at the angular spread given above in cases where small momentum transfers have to be investigated. On the other hand, it is easy to obtain a very good energy resolution by making the Bragg angle sufficiently close to π .

E.g. for $\theta = \pi - 0,1$, $\delta\theta = 1^\circ$ one finds $\delta E/E = 8 \cdot 10^{-4}$! Back diffraction at these very large angles is easily possible because the instrument is situated at the end of a neutron guide where the background from the beam is low.

3.5 COMPARISON

Using the quoted data we find

$$F \delta V_0 \delta V_1 f f_1 e^{-E_0/k_B T} = \Delta R_{\text{crit}}^2 A_0^4 \frac{\beta^2}{4} \left(\frac{\hbar \omega}{E_0} \right)^2 a \cdot \begin{cases} 4 \cdot 10^{-2} 25 f_1 e^{-E/k_B T} (t-o-f) \\ 1,8 \cdot 10^{-3} \cdot 50 \cdot 0,25 e^{-E/k_B T} (\text{triple a}) \end{cases}$$

Therefore, the optimum obtainable intensity is independent of the neutron energy except for the Boltzmann factor which gives a certain advantage with respect to lower energies in the case of a cold source; and it is quadratic in the resolution β .

Using the data cited above, and investigating quasi-elastic scattering ($f_T = 0,25$), the time-of-flight instrument is approximately 10 times better than the triple axis spectrometer with regard to intensity.

For the determination of the dispersion law, the time utilization is poorer and also F is smaller. In this case the triple axis spectrometer will be equal or superior. These conclusions are true

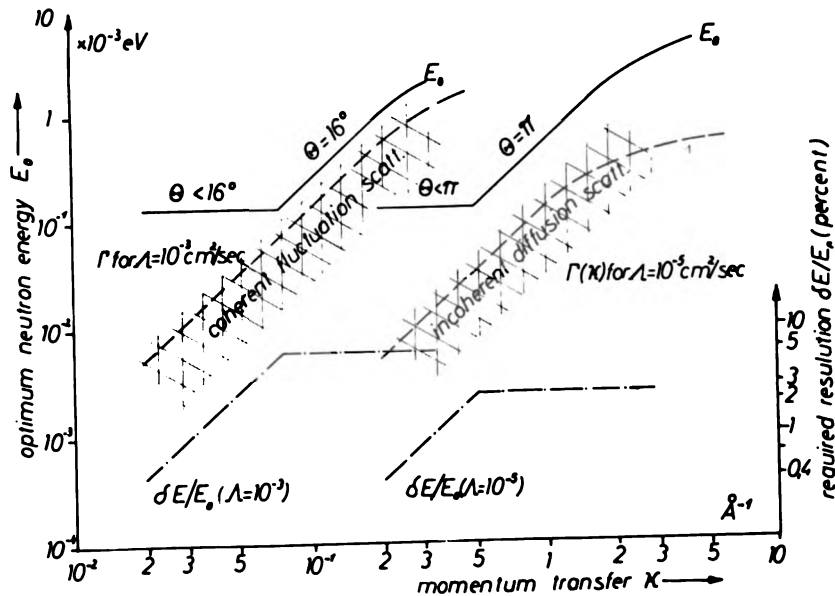


Fig. 3.2 The width of the quasi-elastic line $\Gamma = 2\kappa\Lambda\kappa^2$ with two values of Λ , one typical for incoherent diffusion scattering, the other for coherent fluctuation scattering (dashed lines). The interesting regions for both types of investigations are indicated by shading. The neutron energy E_0 which should be used to achieve optimum conditions is given as a function of the momentum transfer κ which has to be studied. The corresponding (optimum) scattering angle θ is indicated. It is assumed that the resolution of the energy transfer is $\beta = \delta E_0/\Gamma \approx \delta E_0/\Gamma = \delta\kappa\omega/\Gamma = 0.1$ in all cases. The required energy resolution $\delta E_0/E_0$ is also shown. The time utilization is proportional to the distance of the solid and the dashed curve ($f_T \sim \Gamma/E_0$). A lower limit of the energy E_0 has been assumed due to the increase of reflection losses. For the regions where the scattering angle θ is far from π , the solid angles cited in the text have to be reduced to allow the required resolution of 0.1, cited in the text.

as long as back scattering is allowed (and therefore the relatively large solid angles of the time-of-flight instrument).

An inherent advantage of the time-of-flight instrument, as compared to the triple axis spectrometer, is that energy, resolution, and time-utilization can be varied independently and easily. Furthermore, the shape of the volume elements in momentum space is very simple and essentially determined by geometrical factors. The latter point is of importance for the quantitative determination of the scattering law, e.g. of liquids. As already mentioned, a certain restriction of the solid angles at the time-of-flight instrument is due to the costs. The following table gives roughly estimated costs per square meter for counters with a sensitivity between 40 and 70 per cents for thermal neutrons (assuming current German prices):

(1) BF_3 2" double layer	19.000 ¢
(2) He^3 (1/2", single)	110.000 ¢
(3) thin $\text{Li}^6\text{F-ZnS/Ag-}$ scintillator	27.000 ¢

The price for the scintillation counter (3) developed by W. Gläser [12] includes the multiplier and the pulse shaping electronic units.

For thin samples, the time resolution is essentially given by the detector thickness. Therefore, if the costs would be normalized to equal solid angle, the values for (1) and (2) would be approximately equal. Another restriction of large detector areas is due to the background which is considerable ^{also} at large distances from the reactor. Typical values measured at the FRJ-2 (DIDO) reactor are shown in figure 3.3. Therefore, even at the end of long neutron guides a certain amount of shielding is needed.

4. DESIGN STUDY FOR A FISSION PRODUCT SPECTROMETER. AT THE HFR.

4.1 SCOPE OF THE EXPERIMENT

A focussing parabola spectrometer with a resolution of $\frac{M}{\Delta M} \approx 800$ has been designed to extract from an internal fission source a beam of mass separated fission products. The instrument is suited for a wide range of experiments with different scopes.

(a) The production of μC - sources for all the short-lived fission products from thermal neutron fission is of main interest in the investigation of nuclei far off the

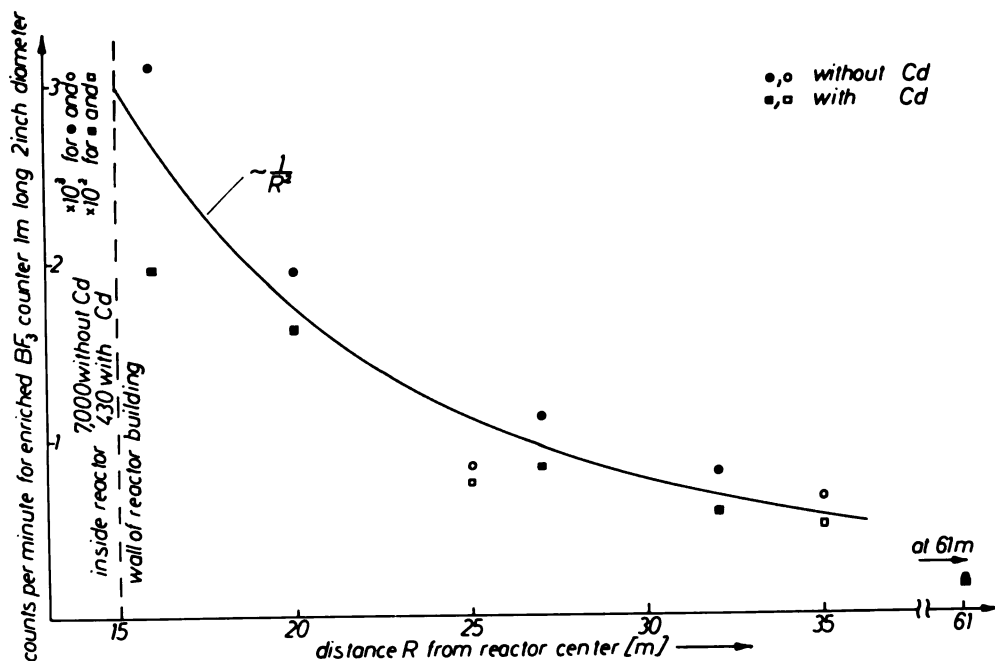


Fig. 3.3 Neutron background outside the FRJ-2 reactor building (reactor power 10 MW) (B. Dorner, private communication). Measurement at the axis of 10 H hole, with 20 cm Be-filter:

- counter without Cd
- counter with 3 mm Cd

Measurements 10 meters apart from the beam axis:

- without Cd
- with Cd;

measurements 5 meters apart are not widely different from the 10 m values.

stability line of nuclei. The number of nuclei to be expected amounts to about 200 with halflives ranging down to 0.5 sec. Nuclear spectroscopy of the neutron rich short-lived nuclei will give valuable information on nuclear structure in unexplored regions of the mass valley. The steepness of the mass energy valley far off the stability line is of major importance in testing the different mass formulas.

- (b) Information on the fission process itself such as the charge distribution of the fission products, and its dependence on mass and excitation energy of the fragments, may be obtained from an investigation of the β - decays of the fragments. The problem of nuclear charge distribution will be finally solved from the experimental side. Special aspects of the distribution of mass and kinetic energy in fission might be studied.
- (c) A beam of fast (100 MeV) radioactive ions is a valuable means in the investigation of the slowing down process of heavy particles. The interaction of fast ions with matter, as Coulomb interaction with nuclei, electronic excitation and lattice effects is a wide field of application.

4.2 CHOICE OF SPECTROMETER

The separation of the fast, unsloved fission products faces the experimenter with two problems unknown to conventional mass spectrometer techniques. [13].

- (a) The ionic charge of the fission products is not well defined but spread about an average value with a FWHM of 25%. Each mass value combines with about 10 ionic charge values.
- (b) In the fission process a spread in the kinetic energy of the fission products of fixed mass is observed. Its value at FWHM amounts to 14%. In order not to loose any particles a fission product mass separator should accept an energy band which is much broader than in available spectrometers.

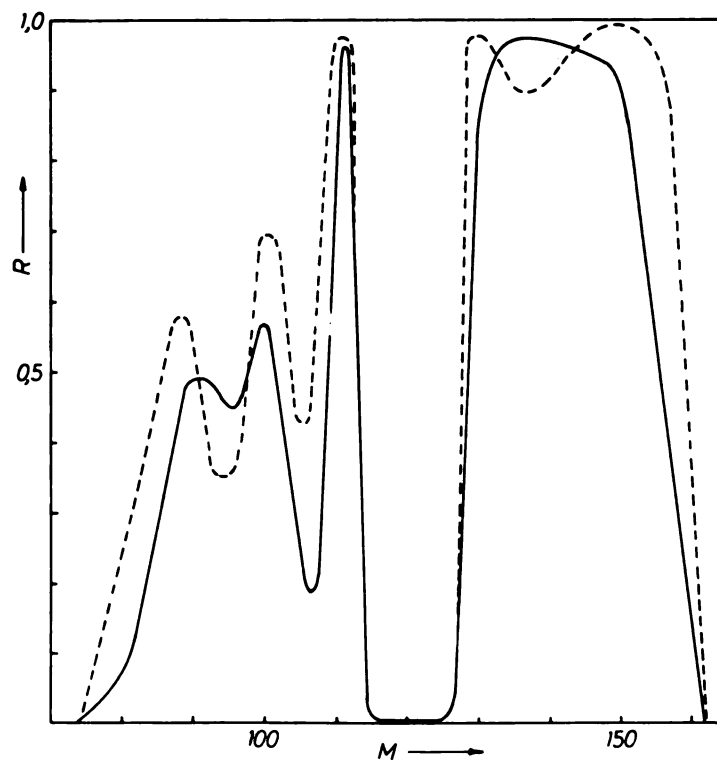


Fig. 4.1 Purity of the fission product beam for $\frac{M}{eA}$ -150-spectrometer

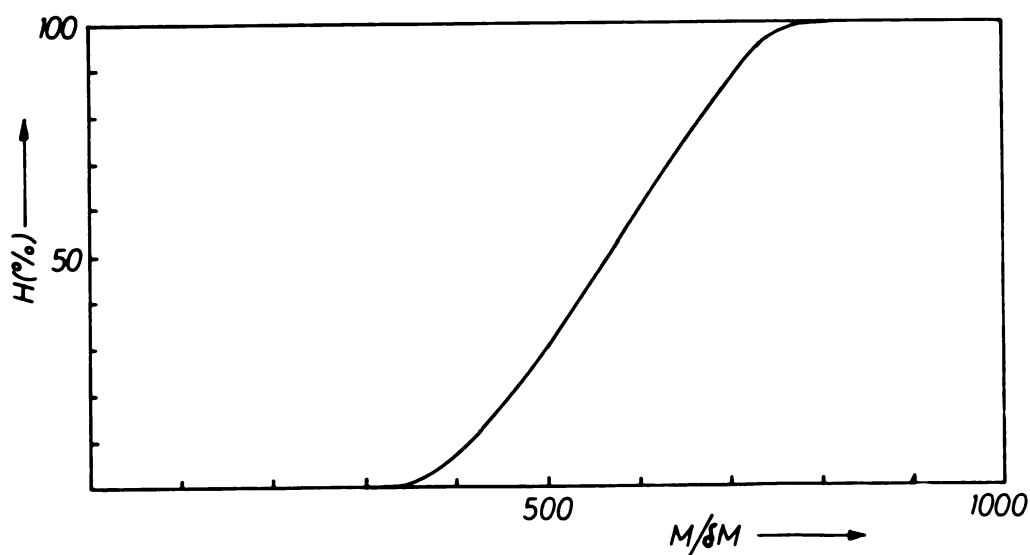


Fig. 4.2 Probability of finding a $\frac{M}{eA}$ -value which gives 99% pure isotopes.

A conventional separator with a mass resolution sufficient to separate masses which differ by one mass unit applied to the analysis of fast fission fragments emerging from a thin layer of fissionable material produces no admixture-free pure isotopes. The admixture caused by overlap of adjacent e/m - values depends on the mass value of the fragments. Fig. 4.1 gives the purity defined by the ratio of the number of wanted to the total number of fission products. The higher purity in the heavy fragment group has been used for fragment separation by Ewald et al. [14] to investigate fission products in the mass range (130 - 145) by a Herzog-Mattauch-spectrometer.

The resolution needed to find for each fragment mass one e/m - value which gives 99% pure isotopes is about $\frac{M}{\Delta M} \approx 800$. Fig. 4.2 gives the percentage of all possible masses which can be separated admixture-free by a separator of given mass resolution.

A mass separator for the production of pure isotopes from fast fission fragments should fulfill the following requirements:
 (1) A mass resolution of $\frac{M}{\Delta M} \sim 800$ and ⁽²⁾ an accepted energy width of at least 10%.

The Herzog-Mattauch-type spectrometer that is a spectrometer with deflection in the magnetic and electric field in the same plane, is velocity focusing and accepts an energy width of 1% or less. In the investigation of single isotopes its main advantage of focusing several adjacent masses can not be used. The energy width of a Herzog-Mattauch-type spectrometer is small compared to the energy spread in fission. If it is used to separate fast fission products, only 9% of the total energy spectrum of the fragments are analysed.

A parabola spectrometer that is a spectrometer with deflection in the magnetic and electric field in perpendicular planes, is not velocity focusing. Particles of definite mass but different energy are focussed on a well defined line which is in the ideal case of parallel fields part of a parabola. For fast fission products a parabola spectrometer has to be built, which focusses stigmatically with the highest possible transmission along a length of the parabola, that corresponds to the energy spread of the fragments.

Until now, only a small focusing parabola spectrometer has been built [15].

An electric and a magnetic sector field with perpendicular planes of deflection have focusing actions in these two perpendicular planes. A definite distance between the fields exists which makes the system of the two fields to a stigmatically focusing system, Fig. 4.3. To achieve a high transmission and stigmatical focusing along the whole energy spectrum the ionoptical aberrations have to be corrected.

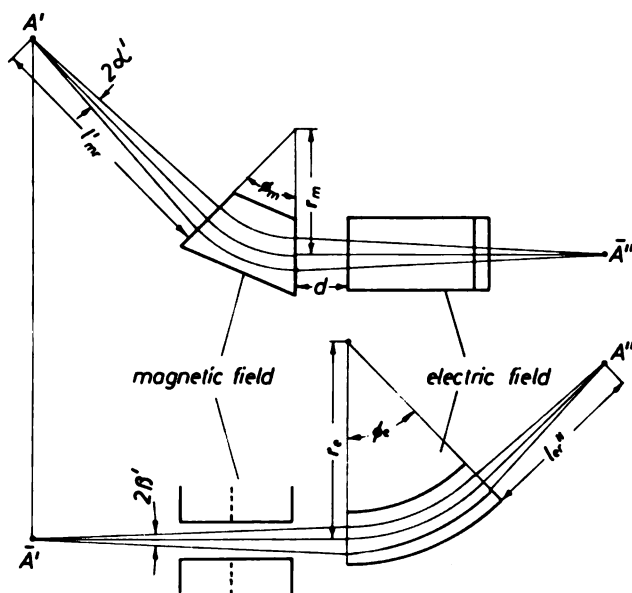


Fig. 4.3 Ion optics and deflection of focusing parabola spectrometer

Very extensive calculations have become necessary to correct the system for second order aberrations. Fringing fields have to be taken into account [16].

The total of all second order aberrations perpendicular to the parabola determine the mass resolution of the system. A machine calculation using the technique of linear programming has been performed to find the system with highest possible luminosity for a given mass resolution and an energy spread of the fragments of at least 10%. A correction of the aberrations caused by the large energy spread and the large angles of divergence is possible by suitable curvatures of the field boundaries [17].

Table 1

Angle of defl. in magn. field	45°	max. magn. field strength	2-3 KG
Angle of defl. in electr. field	42,5°	max. voltage in electr. field	600 KV
distance source-magn.field	700 cm	length of magn. field	275 cm
curvature in magn. field	350 cm	length of electr. field	415 cm
curvature in electr. field	560 cm	length of main path	2030 cm
angle beam axis- normal of field boundary	0		

Table 1 gives as an example a set of data for a focusing system. The luminosity of the system amounts to $12 \cdot 10^{-5} \text{ cm}^2$, if a resolution of $\frac{M}{\Delta M} = 200$ is required. The luminosity decreases to $0,6 \cdot 10^{-5} \text{ cm}^2$, if the resolution is increased to $\frac{M}{\Delta M} = 800$. The area of sources amount for the two above cases to $0,7 \times 7 \text{ cm}^2$ correspondingly to $0,14 \times 3,5 \text{ cm}^2$. With increasing energy width the mass resolution along the parabola decreases, for example from $\frac{M}{\Delta M} = 200$ for $\frac{\Delta E}{E} = 0$ to $\frac{M}{\Delta M} = 150$ for $\frac{\Delta E}{E} = \pm 5\%$. The energy resolution of the system is $\sim 1\%$. The calculations make sure that no combination of homogeneous electric and magnetic sector fields exists with a considerably higher luminosity. The number of separated fragments may be estimated from the known luminosity. The thickness of the fissionable layer should not exceed 400 ugr/cm^2 . A thermal neutron flux of $\Phi = 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$ is required to produce 1 uC-pure sources from maximum yield fission products (6%) with a halflife of 1 sec.

The proposed experiments demand a very high flux reactor as most yields are smaller than 6% and many halflives greater than 1 sec. /18/.

4.3 SPECIAL ASPECTS OF A FOCUSING PARABOLA SPECTROMETER AT THE HIGH FLUX REACTOR

The installation of a parabola spectrometer at a high flux reactor involves some proper design problems.

- (a) In the beam tube a vacuum of at least $1 \cdot 10^{-6} \text{ mm Hg}$ has to be maintained.

- (b) Adequate cooling of the beam tube has to be provided. Burn up of the fissionable material and sputtering of unfissioned atoms from the thin layer of fissionable material restrict the lifetime of the fission sources to a few days. Changing of the highly active fission product sources (10 C) without breaking the vacuum at full reactor power should be provided.
- (c) The fission product gases Kr and Xe have to be safely removed from the vacuum system. A special gas tight system is necessary to prevent the radioactive gases from leaking into the reactor hall.
- (d) No cooling of the fission product sources will be provided. The source holder made of titanium is heated by nuclear heating to about 500°C. Direct thermal contact of the hot source holder and the cooled beam tube should be avoided. Fig.4.4 gives a draft of the planned experimental set up outside the shielding of the high flux reactor. A horizontal, tangential beam hole of 15 cm diameter will be used for the fission product spectrometer. The thermal flux at the source position is $5 \cdot 10^{14}$ n/cm²sec. The sources will be changed without breaking the vacuum and without disturbing the adjustment of the spectrometer from the spectrometer side during full power operation of the reactor. Fig.4.5 gives the in-pile installation.
- The space requirements for the spectrometer outside the reactor shielding are 12 m in the direction of the neutron beam axis, 12 m in the direction perpendicular to the neutron beam axis and 7 m in height. The estimated costs of the installation amount to about 300.000 US \$. The spectrometer will be built by a cooperation of the Institute for Neutron Physics of the Kernforschungsanlage Jülich and the II. Physical Institute of the University Gießen (Prof. Ewald).

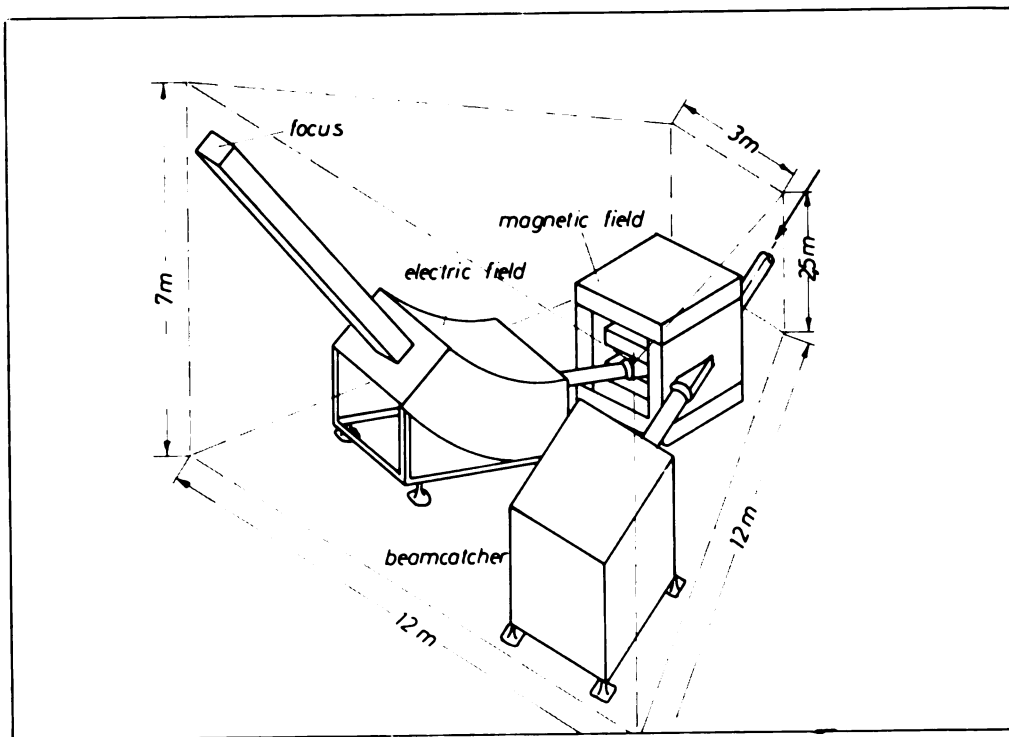


Fig. 4.4 Out-of-pile part of the spectrometer

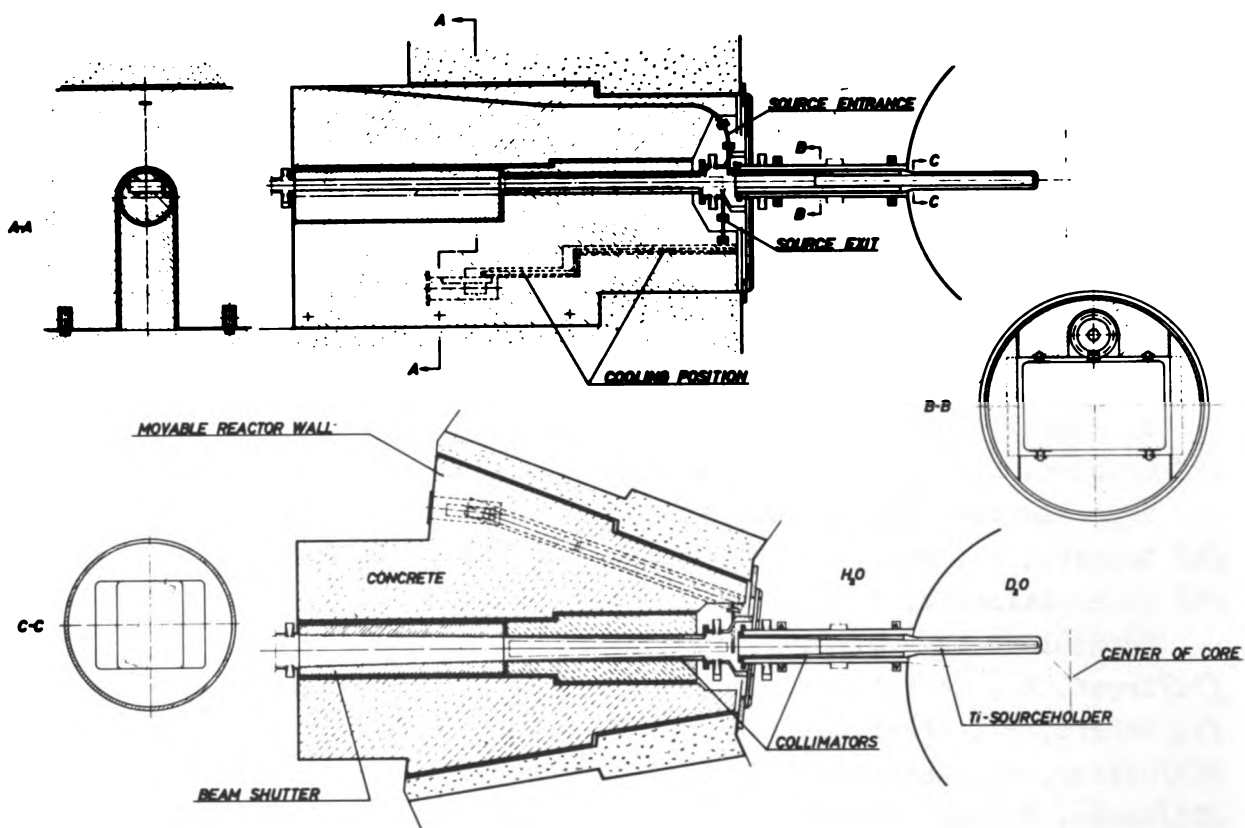


Fig. 4.5 In-pile installations of spectrometer.

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SESSION VI

PANEL DISCUSSIONS ON THE USE AND COMPARATIVE MERITS OF SYSTEMS

SESSION VI PART A - PANEL DISCUSSION

ON

STRUCTURE STUDIES INVOLVING DIFFRACTED NEUTRONS

Panel Members: G. Caglioti, Chairman
R. Brugger
B. Buras
J. Hastings
M. Wilkinson

Secretary: N. Nereson

CAGLIOTI: The following remarks will serve to define what we usually mean by the term diffraction.

In 1954 it was pointed out by Van Hove that the partial differential cross section of a condensed system of particles for the scattering of a neutron beam can be expressed in terms of a scattering function $S(\vec{Q}, \omega)$, which gives the intensity of the diffracted neutron as a function of the wave vector transfer \vec{Q} and energy transfer $\hbar\omega$ undergone by the neutron as a result of the scattering process. The wave vector transfer and the energy transfer are defined in terms of the momentum and kinetic energy of the impinging and final wave vector of the neutron as follows:

$$\vec{Q} = \vec{k}_0 - \vec{k}' \quad \text{and} \quad \pm \hbar\omega = \frac{\hbar^2}{2m} (k_0^2 - k'^2) \quad .$$

$S(\vec{Q}, \omega)$, i.e., the intensity of the neutrons undergoing a momentum transfer $\hbar\vec{Q}$ and an energy transfer $\hbar\omega$, turns out to be the Fourier transform F of the space-time relation function $G(\vec{r}, t)$ where

$$S(\vec{Q}, \omega) = F [G(\vec{r}, t)] \quad .$$

The function $G(\vec{r}, t)$ specifies the number density of nuclei of the system within a certain volume at a position \vec{r} from the origin at time t and thus gives the complete description of the system under study at a microscopic level.

Diffraction people are in general interested in the quantity

$$S(\vec{Q}) = \int S(\vec{Q}, \omega) d\omega \quad .$$

This quantity for most cases of neutron diffraction by solids turns out to be, to a good approximation, equivalent to $S(\vec{Q}, \omega = 0)$ although there may be important exceptions which I will discuss later.

In the case of liquids, a measurement of $S(\vec{Q})$ practically cannot be done, especially with pulsed sources, unless one takes into account the inelasticity of the scattering process. What I have said will justify possible overlaps between

the topics to be discussed in the present Panel and the topics which will be discussed in the following Panel.

Here, we are interested in information relating to the following topics:

- 1) An estimate of the order of magnitude of the flux necessary to solve the problems that neutron diffraction or solid state physicists will face in the near future, and
- 2) A comparison between continuous and pulsed sources.

The above information requires a thorough knowledge of solid state physics, chemistry, or even biology, and of the methods of neutron diffraction.

We are fortunate to have with us on the Panel two outstanding proponents of the conventional method of neutron diffraction, namely Hastings and Wilkinson, and two pioneers in the recently developed time-of-flight (TOF) diffraction, namely Brugger and Buras.

In order to provide information on the topics just stated, it will be desirable to discuss and review separately the methods of conventional diffraction and the methods of TOF diffraction, and then ask what are the best characteristics of the two sources. This discussion should be conducted by keeping in mind quantities like intensity, resolution, reliability of the data, and procedures to obtain structure factors from the raw experimental data, and the advantages and disadvantages of these two general methods for the study of solids and of liquids.

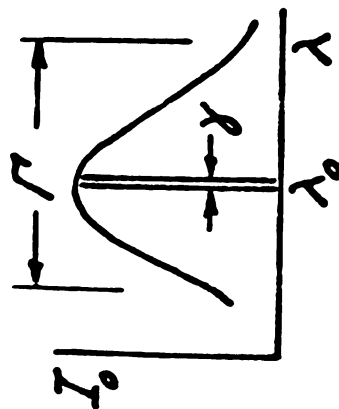
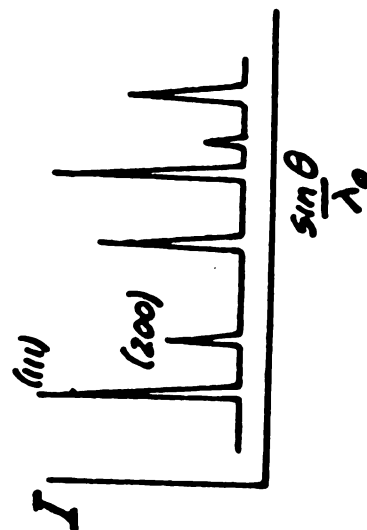
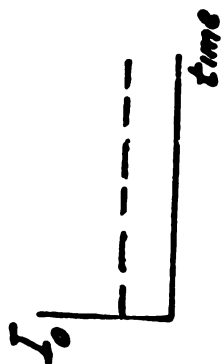
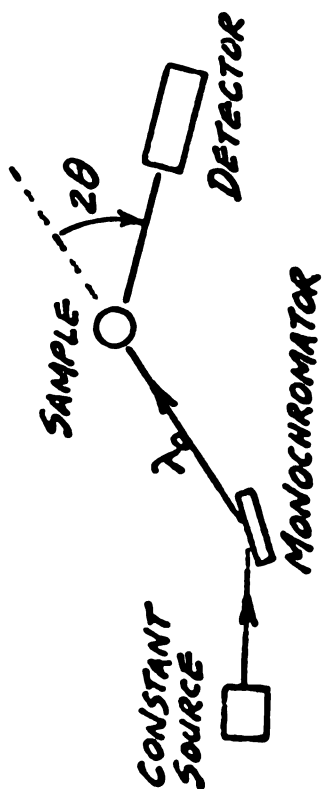
As far as characteristics of the source are concerned, experimentalists know all too well that intensities never appear to be sufficient. In the case of neutron diffraction, the resolution obtainable increases very slowly with the available intensity so that a factor of 10 improvement in the intensity will perhaps result in an improvement of resolution by a factor of the order of 2 for a given counting time. Experimentalists always find some way to improve the quality of their data if they are allowed to use more intense neutron sources.

Now I would like to ask Wilkinson to present his views on the methods of conventional diffraction by the use of continuous sources, and the possible advantages or disadvantages with respect to the methods of TOF diffraction.

WILKINSON: First of all, I would like to state that there is really less difference of opinion among the members of this Panel than one might gather from the discussion that follows. All of us know that there are both advantages and disadvantages in the TOF technique as compared with the conventional techniques for doing neutron diffraction. I will try to explain to you, in an impartial way, the advantages of the conventional techniques.

Let me first focus your attention on the primary differences in these two techniques for neutron diffraction. These differences are illustrated in Fig. 1, which is, incidentally, a figure taken from a paper by Shull (CONF 650217, page 107). In the conventional technique a monochromating crystal is used to obtain a beam of

FIXED WAVELENGTH



FIXED SCATTERING ANGLE

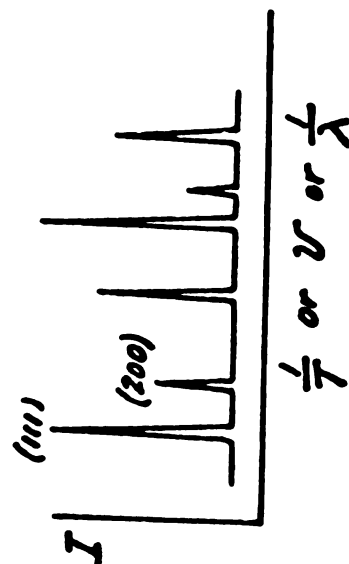
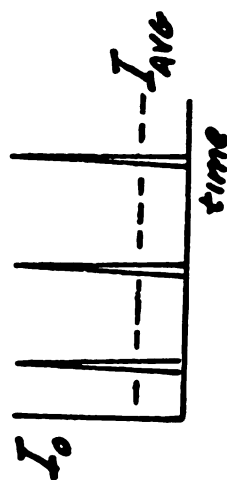
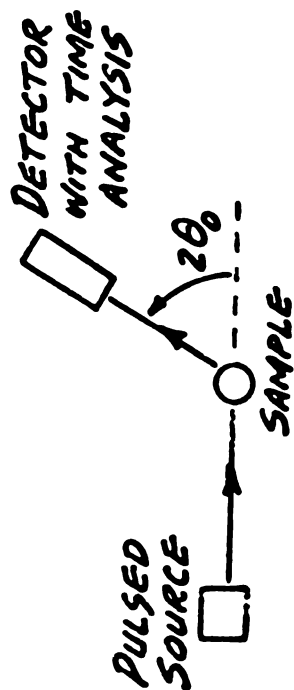


Fig. 1. Comparison of fixed wavelength and fixed angle techniques (from C. Shull, CONF-650217, EANDC-US-74U, INDSWG-76U).

neutrons of fixed wavelength, and these monochromatic neutrons are then incident on the sample. The detector is rotated and the diffracted intensity is measured as a function of the scattering angle. In the TOF technique the source is a pulsed source, which can be obtained either from a pulsed reactor or from a continuous reactor with a chopper. The detector is positioned at a fixed angle and the diffracted intensity is measured as a function of neutron TOF, i.e., energy or wavelength.

There is no doubt in my mind that at the present stage of development the conventional diffraction technique is considerably more accurate than the technique involving TOF. I feel that this will always be the case and there are specific reasons for this opinion. One reason is associated with the corrections that have to be applied to the data. There are corrections, such as absorption and extinction, which must be applied in both cases. The main difference is that in the conventional techniques these corrections are applied at one wavelength, whereas in the TOF technique they must be applied as a function of wavelength, which is considerably more difficult. Another factor, which is very important for accurate intensity measurements, is a knowledge of the intensity of the beam that is incident on the sample. In the conventional technique it is very easy to obtain a precise measurement of the intensity of primary radiation incident on the sample. This can be done by replacing the sample with a calibrating material with well-known scattering characteristics. I do not know of any similar way that such a precise calibration can be achieved with the TOF technique, because the calibration must be made as a function of wavelength. Therefore, if very precise measurements are desired (particularly measurements that require placing the data on an absolute scale), the conventional technique is the better method.

The topic of crystal structure analysis should be examined next, because this is one of the main considerations for having high intensity neutron sources. We all realize the importance of crystal structure determinations and of the use of neutrons in studying complicated organic and inorganic structures. At the present time I am not convinced that it is possible to study complicated structures by the TOF method. In principle, this is possible. In practice, the technique has not been developed sufficiently to prove this capability. Of course, the TOF technique is relatively new, and it has not reached the stage of development of the conventional techniques. Therefore, I am not stating that such work cannot be done; I merely state that at present there is no proof that it can be done. On the other hand, the conventional techniques have been used to determine very complex structures. Complex organic and inorganic structures have been studied by Levy and his group at Oak Ridge, and similar work has been done on some of the simpler biological molecules at Harwell.

Let us assume for this discussion that it will be possible to develop the TOF technique so that complex crystal structure determinations can be made. With this assumption, if a pulsed reactor is the source of neutrons, then there is little doubt that the TOF method should be used. On the other hand, if there is a continuous

neutron source, the situation is different. The reason for this difference is that, for TOF measurements with a continuous source, the beam must be chopped and at least 99% of the incident radiation is discarded. Of course, in the conventional technique the monochromating process also picks out about 1% of the neutrons, as indicated in the center diagram of Fig. 1. Therefore, the neutron intensity incident on the sample in the two methods is roughly the same. However, the efficiency of the two experiments is the same only if both make use of the same fraction of neutrons incident on the sample. In the conventional technique the entire beam of incident monochromatic radiation is used. In the pulsed technique, if the sample is polycrystalline, crystal planes positioned in different orientations scatter from the pulsed white beam the approximate wavelengths for a fixed scattering angle. Consequently, all of the incident neutrons are effectively used, and the two techniques are comparable. For complicated crystal structure analysis, however, it is necessary to use single crystals, and in these investigations the efficiencies of the two methods are no longer similar. With a single crystal, only certain planes are oriented to diffract neutrons in the direction of the detector. Therefore, in the TOF method, the single crystal sample acts as a monochromator, and only a small fraction of the beam, which is incident on the sample, is used. The conventional technique will, therefore, allow much faster data collection than the TOF technique.

In order to make up for the above deficiency, Buras has pointed out that a detector which covers a wide range in scattering angle can be used. It is then possible to have several reflections which will give diffraction peaks simultaneously in the detector. As I mentioned in a previous discussion, it is also possible to cover a wide range of scattering angles with the conventional techniques. One obvious method would be the brute force method of using many small detectors. However, it would be desirable to have a large position-sensitive detector which gives accurate intensity measurements over all scattering angles. There are at least two groups that are actively working on detectors of this type. There is one at Brookhaven, which is studying the feasibility of using spark detectors for this purpose; there is one at Oak Ridge, which is interested in using neutron scintillators and image intensifiers. Without speculating on the best method, I merely want to emphasize that the range of detection in scattering space can be increased in the conventional method also. If monochromatic radiation is incident on a single crystal, only very few planes will be properly oriented to diffract neutrons. Therefore, it might not appear that wide angle detection would be of any significant benefit. However, the extra range of detection does help if the crystal is oscillated. This is a time-proved technique in x-ray crystallography, which has been used for many years, and it is a minor extension of the conventional technique of neutron diffraction. This method allows very precise measurements and neutron crystallographers at Oak Ridge are presently planning to use it in structure determinations.

Flight Path = 208 cm.
 Scattering Angle = 90°
 Period Of Chopper = $2140 \mu\text{sec}$
 Aluminum - \circ
 Chromium - \bullet

a - Chopper Burst
 b - Bragg Angle
 c - Flight Path
 d - Length Of Counter
 e - Channel Width ($4 \mu\text{sec}$)

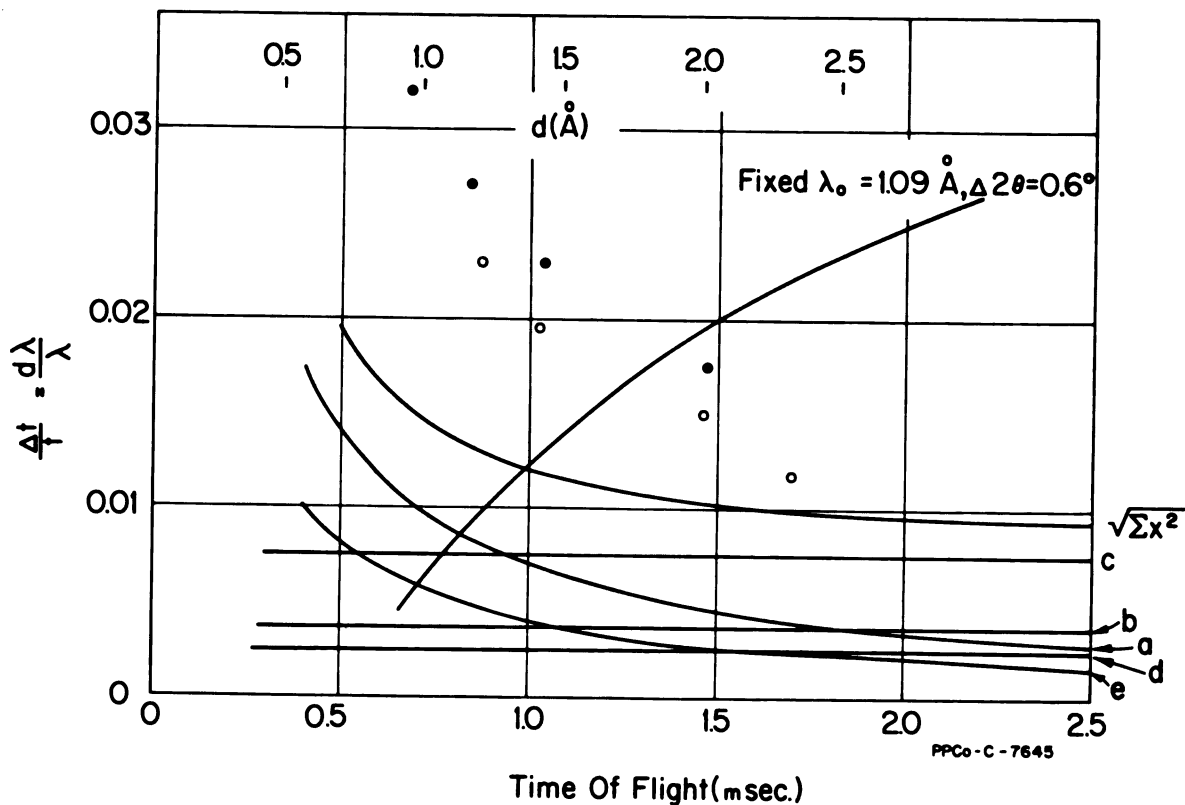


Fig. 2. Resolution of fixed wavelength and Bragg's TOF experiment.

Actually, if the main interest is the speed of data collection, then a detector which covers all of scattering space should be used and, instead of monochromatic radiation, white radiation should be incident on the crystal. This is the old technique which was used many years ago in x-ray work by Laue. In this method all of the neutron radiation is used; there is a continuous source and continuous detection. One disadvantage of this technique is that, at any given angle in scattering space, all possible orders of a Bragg reflection will occur simultaneously. This could present difficulties in the data analysis. If it does, I believe that the technique can be modified slightly by using crude velocity selectors in the incident beam. In this way finite continuous bands of wavelengths, which do not include higher orders, could be selected. I think this selection of bands could be done with crude velocity selectors or with balanced filters. I merely mention this possibility for speculation, because I do not know that the techniques can be developed. Much depends on how well the detection system gives adequate angular resolution throughout scattering space. However, it is a technique that should be considered, because data from a crystal could be collected at least a factor of 10 faster than they could be obtained by other methods.

Let us next consider the question of resolution in these two techniques. A comparison of the wavelength resolution is shown in Fig. 2, which is taken from a paper by Brugger. The curve labeled $\sqrt{\Sigma x^2}$ is supposed to represent the summation of the resolution functions in his TOF chopper system. The actual data points do not fit this curve, but the reason for the discrepancy is known. The uppermost curve in Fig. 2 represents the wavelength resolution for the conventional technique, employing a wavelength of 1.09 Å. Buras showed a similar curve at a previous session, and he indicated methods by which resolution in the TOF technique could be improved. It is also possible to improve the resolution in the conventional technique. It is merely necessary to employ monochromatic radiation of a wavelength longer than the 1.09 Å shown in the figure; the resolution function can be pushed to a much lower level. The problem of contamination by higher order reflections does not appear to be serious in the conventional technique, even with primary radiation of long wavelengths. Monochromating crystals of germanium have been used at Brookhaven, and I understand that the second order contamination is extremely small. Furthermore, it is always possible to use a very crude velocity selector. Velocity selectors can be obtained that allow about 80% transmission at a given wavelength and effectively zero transmission at the higher orders. Therefore, I do not think that the question of higher order contamination is a serious problem for either technique.

The reason that most conventional techniques have used neutron wavelengths of the order of 1 Å is primarily because there are more such neutrons available from the reactor and because most experiments have not required high resolution in the part of the pattern represented by large d spacings. It is usually more important to have high resolution in the region of the pattern with small d spacings. If higher resolu-

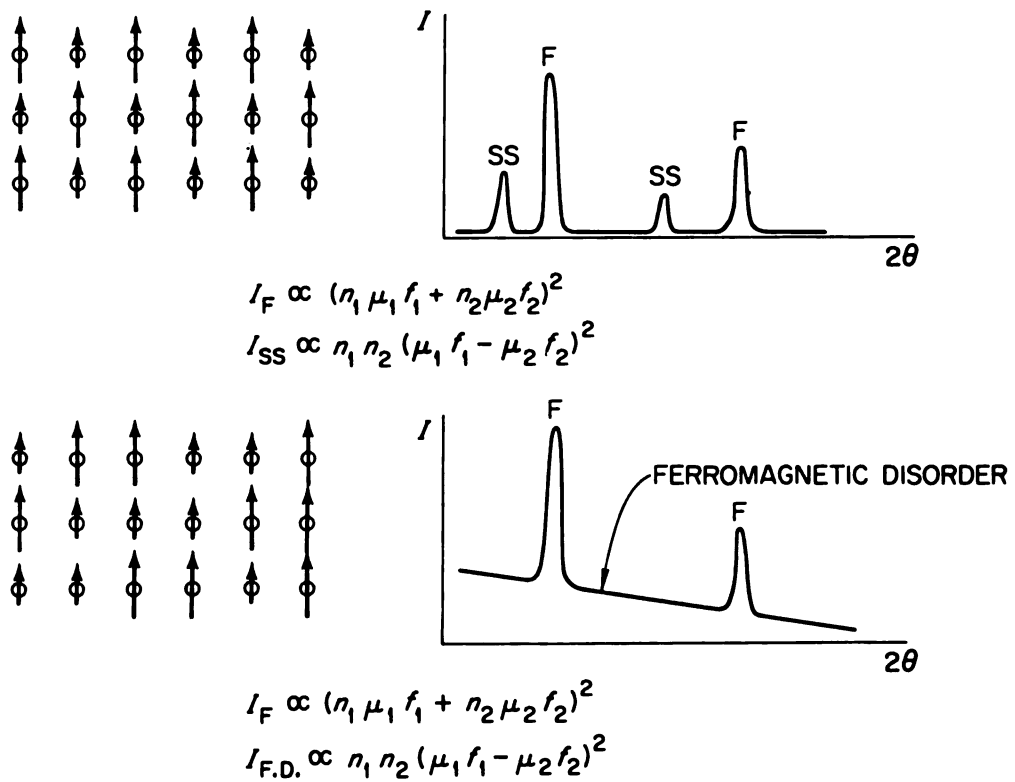


Fig. 3
Neutron Diffraction from Ferromagnetic Binary Alloys.

$$I_{hkl} \propto (b^2 + 2bp \vec{q} \cdot \vec{\lambda} + q^2 p^2)_{hkl} R_{hkl}^2$$

$$\text{where: } \vec{q} = \vec{k} - \vec{\epsilon} \quad (\vec{\epsilon} \cdot \vec{k})$$

$$q^2 = 1 - (\vec{\epsilon} \cdot \vec{k})^2$$

with $\vec{H} \perp \vec{\epsilon}$:

$$\vec{\epsilon} \cdot \vec{k} = 0; \quad \vec{q} \cdot \vec{\lambda} = \pm 1; \quad q^2 = 1$$

$$I_{+hkl} \propto (b + p)_{hkl}^2 R_{hkl}^2$$

$$I_{-hkl} \propto (b - p)_{hkl}^2 R_{hkl}^2$$

Fig. 4 POLARIZATION OF NEUTRONS
BY BRAGG REFLECTION

tion is needed to study certain properties of large unit cells, e.g., magnetic unit cells, it is always possible to utilize monochromatic radiation of longer wavelength. However, since there is a smaller fraction of such neutrons available from the reactor, the conventional techniques have not used the longer wavelengths except in specific problems.

It appears to me that the region of small d spacings is the region for concern about resolution in the TOF technique. I do not know how much the resolution can be improved, but I am convinced that for complicated structure work, e.g., investigations of biological molecules, it is necessary to resolve and to precisely measure the intensities of the higher-index reflections. These reflections will occur in diffraction patterns at d spacings of 1 \AA and below. Since the resolution function increases sharply for low d spacings, I doubt at the present time that the TOF technique can be improved so that it can be used for studying very complicated structures.

I shall now turn to a couple of classes of problems where I feel that the conventional techniques offer considerable advantages over the TOF method. The first class involves those problems where it is necessary to obtain a precise measurement of elastic scattering, which occurs as part of the diffuse scattering. An illustration of a problem of this type is shown in Fig. 3, which is a schematic diagram of diffraction patterns which result from ferromagnetic binary alloy systems. If the magnetic moments are ordered in position within the specimen, then additional diffraction lines, referred to as super-structure (SS) lines, occur in the diffraction pattern. If these moments are completely randomly located in position, the super-structure lines do not appear and the magnetic scattering is part of the diffuse scattering. In such cases, it is necessary to make precise measurements of this ferromagnetic disorder scattering. Such measurements can very easily be made in the conventional technique and it is easy to put these measurements on an absolute scale. Information is obtained from the measurements on the absolute magnitude of the magnetic moments in the system and on the spatial distribution of the magnetic electrons. These are very important results, which are necessary in understanding the magnetic properties of alloy systems. I do not see how similar measurements can be made with the normal TOF method.

Perhaps an even more important class of experiments that will be difficult or impossible for the TOF technique is the use of polarized neutrons. I think many of you are familiar with the method which is used in obtaining polarized neutrons in the conventional experiments, and this method can be illustrated very briefly by the equations that appear in Fig. 4. If a magnetic material has atomic magnetic moments, then the scattered neutron intensity in a particular hkl reflection is given by the quantity I_{hkl} . In the expression for I_{hkl} , b is the nuclear scattering amplitude, p is the magnetic scattering amplitude, q is a vector relating the unit vector \vec{n} (which represents the orientation of the magnetic moments) and the unit scattering vector \vec{e} , $\vec{\lambda}$ is a unit vector representing the spin state of the neutron,

and R_{hkl} is a geometrical structure factor. If a magnetic field is applied in a direction perpendicular to the unit scattering vector \vec{c} and the magnetic moments are rotated into this direction, then the scattered neutrons are resolved into two polarization states. The intensities of these states are indicated by the quantities labeled I_{+hkl} and I_{-hkl} . Observe that one intensity is proportional to the square of the sum of the nuclear and magnetic scattering amplitudes, while the other is proportional to the square of the difference. Obviously, if the two amplitudes b and p can be made equal, then a 100% polarized beam can be obtained. This technique has already been used in many very important experiments. Since a polarized beam is obtained by scattering from a crystal, the polarization is accomplished simultaneously with monochromatization. I do not know of any techniques for the TOF method, which furnish such a high degree of polarization so easily and also give a polarized beam with a high neutron intensity.

In conclusion, I would like to state again that there are obviously some specific types of problems where the TOF method is better than the conventional technique. However, for most of the problems that are currently being pursued, I do not think that the TOF technique is competitive with conventional techniques, if the source is a continuous neutron source. Furthermore, I do not believe that the technique was started with the thought of competing with conventional techniques using continuous neutron sources. I am sure that Buras has considered this technique primarily for pulsed neutron sources, and he deserves a lot of credit for developing it. If it can be used for investigations of complicated structures, then I believe it could become a very powerful tool.

CAGLIOTI: The TOF technique is a newcomer in this field and it comes at a time when conventional techniques are already 20 years old. I believe that there a number of points in Wilkinson's talk that merit very serious consideration; perhaps I could comment later on some of these points. Also, I could show some ways to improve the TOF diffraction technique in complicated structures, but first I would like to ask Buras to present his comments.

BURAS: I could make my talk very short by simply saying that I agree with most of the points mentioned by Wilkinson with one exception. This is a small point which I did not discuss when I presented the resolution for the double-axis crystal spectrometer and TOF methods; the resolution in both methods can of course be improved.

Now I would like to elaborate a bit on what Wilkinson has said about the comparison between the two methods. First, consider the problem of absorption. It is true that in the conventional method one has a fixed wavelength. Therefore, a calculation of the absorption is very easy since only geometrical factors are involved; one has no wavelength dependence on the absorption. This is a disadvantage in the TOF method, and I would like to illustrate how we have tried to check on how this could be taken into account. The work was done by my colleagues at Dubna, and Fig. 5 shows a diffraction

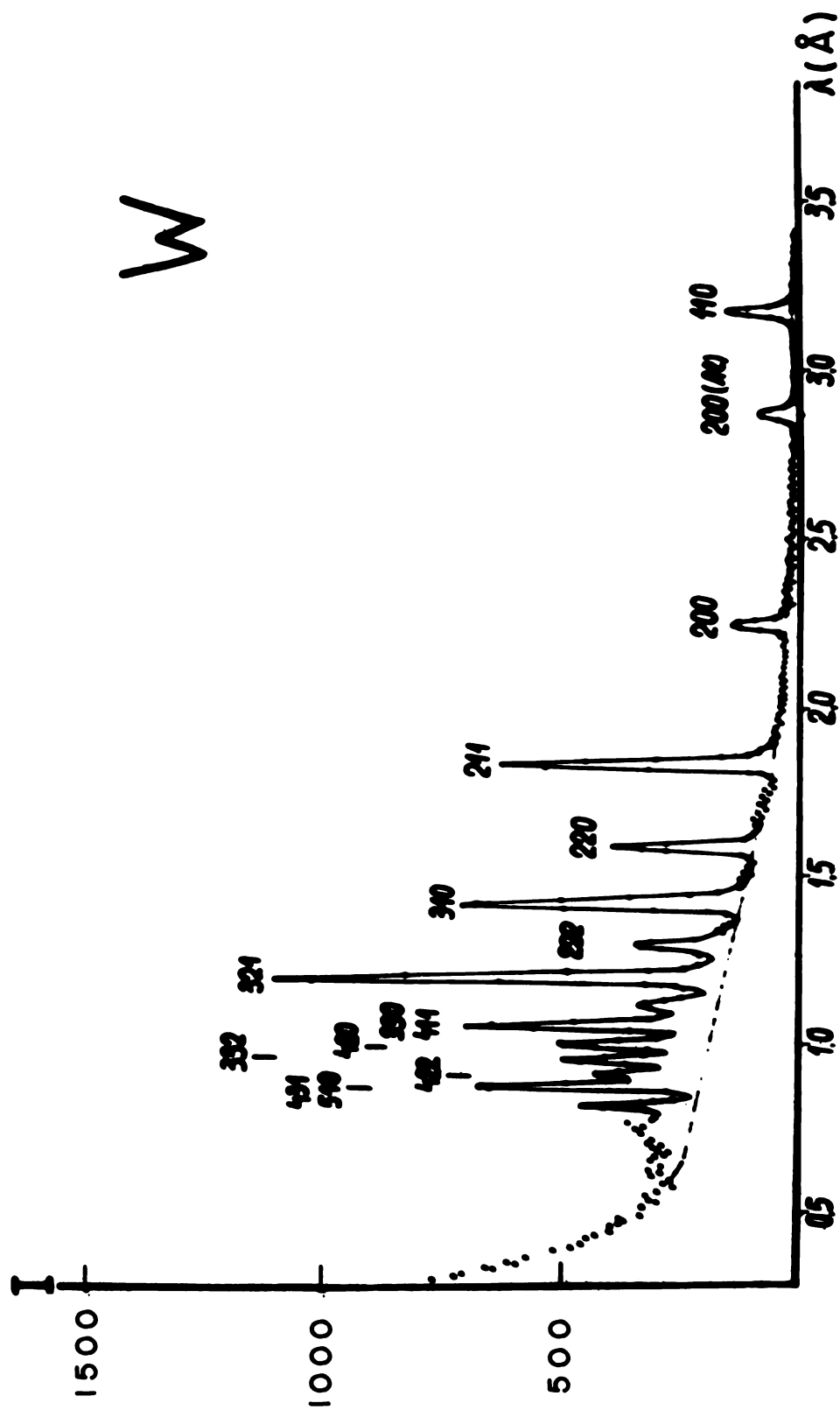
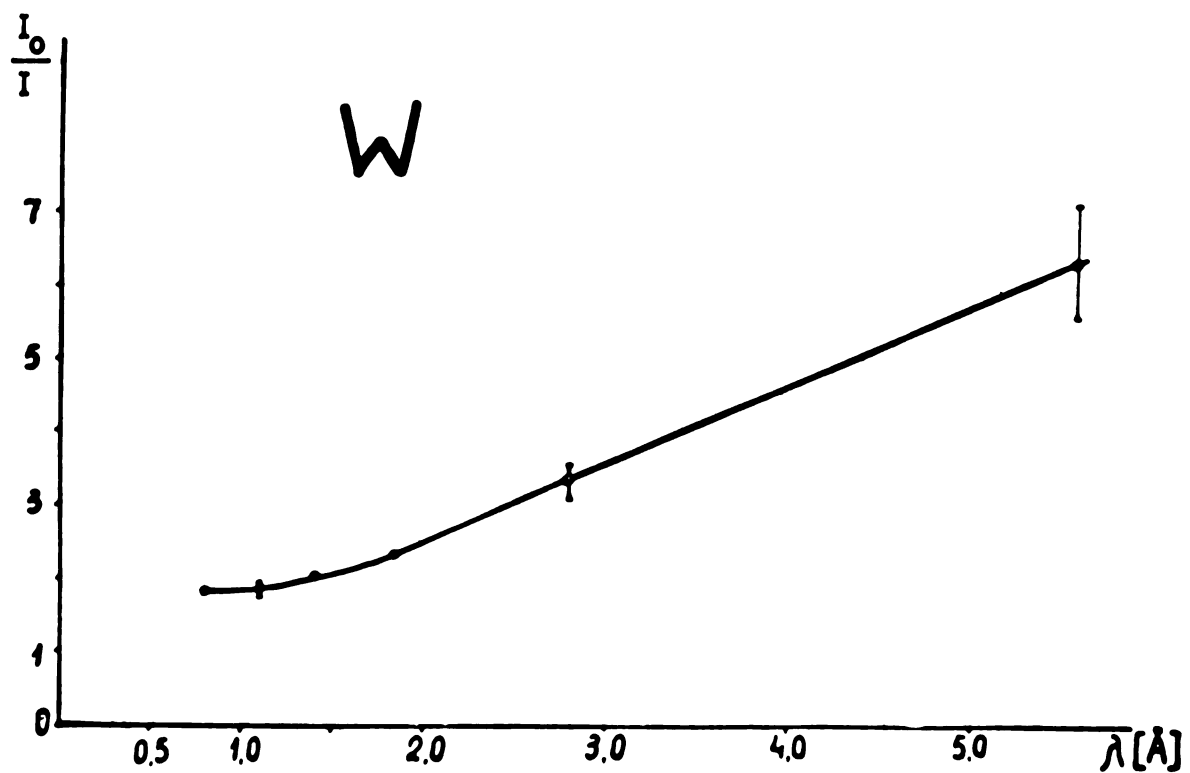
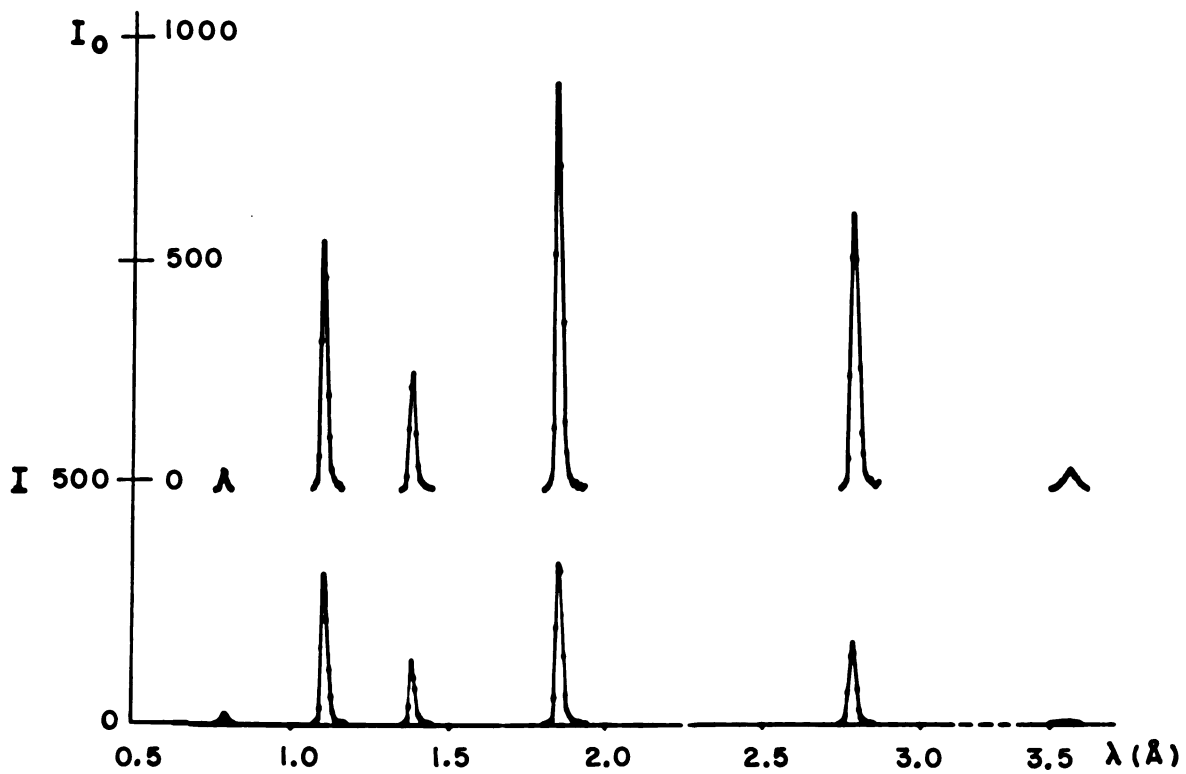


Fig. 5. Diffraction pattern for tungsten observed at Dubna.



Figs. 6 and 7. Absorption caused by tungsten sample and resulting attenuation coefficients.

pattern for tungsten with appreciable absorption. In order to correct for the absorption we need to know the absorption cross section. One could simply obtain this value from BNL-325 but one has difficulty with these values partly because one has a sample in which the density and other quantities characterizing it are not known very accurately. What we have done is to use a bismuth single crystal for the absorption measurements and utilize the higher order reflections at long wavelengths. Fig. 6 shows the effect of absorption on the diffraction peaks using the procedure of sample out and sample in; therefore, for various wavelengths we have simply measured the absorption coefficient, i.e., the relative absorption coefficient for this sample. The results of these absorption measurements are shown in Fig. 7 where we have plotted the wavelength dependent absorption measured for tungsten.

Table I shows, for certain planes, calculated values of jF^2 and experimental values of jF^2 with and without the absorption correction; note that the agreement between calculated and experimental values improves from 23% to 8% when the absorption correction is taken into account. These are not very accurate data but are presented to demonstrate, in principle, that the measurement can be done in this manner.

Table I

hkl	λ	jF^2_{calc}	jF^2_{exp}	Δ	R	$A^{-1}(\lambda)$	jF^2_{aexp}	Δ	R_a
110	3.170	10.22	4.49	-5.7		0.155	9.03	-1.19	
200	2.241	5.00	3.66	-1.3		0.243	4.69	-0.31	
211	1.830	19.62	19.88	+0.3	23%	0.312	19.88	+0.26	8%
220	1.585	9.61	9.54	-0.1		0.354	8.41	-1.20	
310	1.418	18.85	25.85	+7.0		0.380	21.21	+2.36	

The second problem is extinction and I must immediately agree that the problem of extinction in the TOF method is much more difficult. I have shown in previous sessions some calculations based on the Monte Carlo method, and I do not want to discuss this problem again, but I hope that in time this problem can be accurately solved.

Another disadvantage of the TOF method is the problem of the incident spectrum. In order to calculate the structure factors, one must know the incident spectrum. This can be accomplished in two ways. What we have done is to measure the incident spectrum directly with the same counter as the diffracted beam; if done in this manner, one does not need any additional corrections for the counter efficiency. However, one could also do it by calculating the spectrum from diffraction patterns of known crystal structure factors. Curve (a) in Fig. 14 of my paper is the incident measured spectrum, $I(\lambda)$, from the IBR reactor. Curve (b) is the "effective" spectrum, $I(\lambda) \times \lambda^4$, as calculated from curve (a), and this "effective spectrum" is used in the calculation of the structure factors. This is of course a very rough sketch so there are some kinks in the calculated curve; it should, of course, be a smooth curve. The

points surrounding curve (b) were obtained by calculating the "effective" spectrum from diffraction patterns of silicon powder. Some of the points were observed at a scattering angle of 52° , and others were obtained at an angle of 90° . We can use different angles and in this way one can simply measure the whole effective spectrum and solve the problem; however, I would agree immediately that at the present time this is a less accurate method than the conventional one and this should be clearly stated.

Now let us come to the intensity problem. As was said before, if we use a monochromatic beam in one case and in the other case use a chopper at a steady-state reactor the intensity is more or less the same because, as Wilkinson pointed out, in one case we have the chopper and in the other case we have a monochromator. I should say at this point, and it was also pointed out by Wilkinson, if we have a pulsed source this, of course, immediately gives a great advantage to the TOF method because one is not losing intensity by chopping the beam. But let us talk about a steady-state reactor with a chopper. In the conventional method, experiments utilize only one detector. One could of course, as Wilkinson said, improve the method with many detectors and in this way increase the intensity or, in other words, use the entire available intensity; however using a double crystal spectrometer this is very complicated. As I state in my paper, this could be done in the TOF method by using several detectors or, in principle, a 4π detector. There is, as yet, no experience in this matter in the conventional method. We have limited experience with multi-detectors in the TOF method and I have shown several neutron patterns which I think illustrate that this development is possible. And even if in both methods we use, in one case, a 4π counter and, in the other case, several counters, there is one big difference. is supposed to have as many counters (as I understand it) corresponding to as many reflections as one would like to measure.

With conventional techniques one has the problem of varying geometry. In the TOF method, one has no moving part in the arrangement at all unless one is using a chopper and a steady-state reactor. What one does is as follows: The sample is positioned, the time analyzer is started, and one measures the whole spectrum. It may be an additional advantage that one may measure all peaks simultaneously, not simply one by one, but all at essentially the same time. I ought to repeat here that the Laue method with a white beam suffers from the disadvantage of higher order contamination. This could be solved, as Wilkinson proposed, by a crude type of chopper but it complicates the apparatus; with the TOF method one has no such difficulty. In addition, one has some advantage because a certain plane gives not only one reflection, but several higher order reflections which means that better accuracy is obtained in the measurement of a given structure factor.

Now let us consider the problem of resolution once more. The information presented in Fig. 16 of my paper is similar to Fig. 2 as shown by Wilkinson from the paper by

Brugger. Here is plotted the resolution, $\Delta\lambda/\lambda$, as a function of wavelength and the lattice spacing. Data are presented on the resolution for the TOF method at different chopper speeds and also for the conventional method. Note that by chopping the beam with a proper chopper velocity one can achieve a resolution of 1-1/2% at lattice spacings of 3 to 4 Å. Also shown are two curves for the IBR reactor using two moderators, one unpoisoned and the other poisoned with 1% H_3BO_3 ; observe that 1% resolution is obtained in the case of the H_3BO_3 poisoned moderator.

I will agree that for polarized neutrons and diffuse scattering, as it stands now, you cannot use the TOF method, but I do think that in the course of time the features of this method can most probably be improved. This is a very young method and I think that the greatest advantage of the method lies not in conventional uses for this technique, but rather in special studies such as transient phenomena or involving the use of high external fields, e.g., pressures and so on.

BRUGGER: I would like to direct my comments in a slightly different direction than have Wilkinson and Buras. I think it is obvious that neutrons complement x rays because x rays have a 30-year lead time on neutrons. In the same way, the TOF methods for structure work will complement the steady-state methods, and I think that the method needs to be further developed.

In session I.A. Kouts asked the question, "If another research reactor is to be built, what would the experimentalists like to see in this reactor?" Briefly, I would like to try and answer this question. The tabulation below outlines what I would like to have in such a reactor. The underlined values are the more important or desirable portion of each range.

1. Energy range	1 \rightarrow (<u>10-100</u>) \rightarrow 1000 meV
2. Energy adjustable	1 \leftrightarrow 1000 meV
3. Burst time	1 \rightarrow (<u>5-10</u>) \rightarrow 50 μ sec
4. Rate	5 \leftrightarrow 50 pulses/sec
5. Burst	No chopper
6. Beam radiator size	\sim 10 cm x 10 cm and adjustable
7. Source	$\geq 10^{16}$ n/cm ² sec (steady-state reactor equivalent)
8. Background	Low
9. Shield thickness	< 5 m
10. Number of beam tubes	5-15
11. Flight path	5-100 m
12. Detectors	Large area

I have selected this set of reactor features because my experience is with the TOF method and I wish to continue using this method. I think the method has a very beneficial place in the study of solids, and I think that possibly a high flux reactor is easier to obtain on a pulsed basis than on a steady-state basis. The energy range

of a thermal reactor neutron spectrum is approximately from 10 to 100 meV. It is possible to extend the range down to 1 meV and perhaps up to 1000 meV. Above 1000 meV, I see very few experiments to perform. The possibility of going below 1 meV presents difficulties so I place the lower limits of the energy range at 1 meV. It would be desirable to be able to adjust the peak of the spectrum to gain intensity. This can be done by using cold sources to decrease the peak to about 1 meV.

In the proposed reactor I would like to see burst times between 5 and 10 μ sec. I doubt that I would use efficiently pulses shorter than 1 μ sec while pulses longer than 50 μ sec would be very disappointing. Unfortunately, I expect that the pulse widths will probably emerge at about 30 μ sec. The burst rate should be between 5 and 50 pulses/sec. This is to accommodate long flight paths and slow neutrons. I would prefer not to use a chopper to obtain a short burst time, because choppers exclude other beam holes; they also add one more mechanical source of trouble. The beam radiator size should be of the order of 10 cm by 10 cm. This size fits the flight paths and the intensities that I would hope to use; it would also be advantageous to be able to adjust the angle of the source or radiator so that one could use focusing as Buras has pointed out.

The source flux at the face of the moderator should be equivalent to or greater than a steady-state reactor of 10^{16} n/cm²sec. If I evaluate the IBR reactor and the RPI accelerator correctly, these are about equivalent to a 10^{14} n/cm²sec steady-state reactor. Thus it would be desirable to increase the flux by two orders of magnitude. Backgrounds from the source should be low. It would be desirable to remove the sources of background; this is being partially accomplished with a pulsed source. If there are delayed neutron precursors, it is an added advantage to remove them. In addition, it would be desirable to suppress the gamma rays because they can be a source of background.

The shielding should be less than 5 m thick because I would like to have flight paths from 5 to 100 m. If the flight path is less than 5 m, it would be difficult to take advantage of the large source area; for flight paths greater than 100 m, the problems associated with equipment, counters, and flight paths become great.

Another important feature that somebody hopefully will provide is a large area detector that is both space and TOF sensitive so that we can really use all of the available neutrons. I think it will be easier to produce a space and TOF sensitive detector than it will be to produce a space and energy sensitive detector. Of course, this detector could be used both for the structure and for the inelastic scattering experiments.

What are the major difficulties in these requested reactor design features? The first would be the burst time of 5 to 10 μ sec. At the present time it appears that the reactor engineers will have difficulty in providing 5- μ sec bursts if they have to moderate the neutrons. Hopefully with some development this can be achieved. A

source flux greater than 10^{16} would be a second difficulty. The third difficulty, as I see it, is how to make the above-mentioned detector.

HASTINGS: I have been asked to comment on the design requirements of steady-state reactors from the point of view of an experimentalist interested in elastic scattering phenomena. I have two very brief observations to make. The present generation of 10^{15} flux reactors has just arrived and there is very little experience on which to base any guesses. At the moment it would appear that there are few, if any, experiments dealing with elastic scattering which cannot be tackled with a 10^{15} flux reactor. It is my belief that perhaps a decade will be needed to exploit these new reactors. The experience gained during this time may make further increases desirable. The second comment deals with the magnitude of these increases. The history of research reactor development has shown that new horizons are opened for the experimentalist only when a relatively large jump in flux is made, say a factor of 10. In the best of all possible worlds I would like to see a steady-state reactor with an available thermal flux of $> 10^{16}$ in about ten years.

MAIER-LEIBNITZ: What accuracy do you want in F_{hkl}^2 , the square of the structure factor in structure work?

HASTINGS: Once again, I think that the answer to that depends upon the type of problem. If one is studying an organic molecule and one from which you can gather 10,000 reflections, then the accuracy of any one reflection measurement need not be greater than about 25%. If one is working on a simple system in which the total number of reflections available in reciprocal space is limited, then obviously one wants to measure F^2 with an accuracy of 2 or 3% if possible.

MAIER-LEIBNITZ: Could you tell us whether the Laue method without separation of higher orders would be sufficient for structure analysis in organic crystals using a more or less white beam?

WILKINSON: I would say probably not. The reason I say this is based by the fact that even for the simple biological molecules, one has to be able to study some 25,000 reflections. It is difficult to see how there could be this many reflections without a fair number of higher order reflections. I would not like to make a definite statement one way or the other. I think there are probably some materials where the higher order reflections in certain directions are relatively insensitive to the structure determination, but I do not believe that one can make a valid generalization on this question.

MAIER-LEIBNITZ: More generally, I should like to say that I feel that the problem of intensity and resolution has not been fully treated here nor in the written papers. I think one should never separate the treatment of intensity and resolution and that means one should optimize the equipment and then observe what intensity is

obtained. Using the above procedure, I am sure the curves will look very much alike for both methods. Basically, the intensity from a continuous and a pulsed reactor is not very different with equal peak flux, equal detection angle, and equal resolution.

In diffraction from single crystals, high resolution in neutron velocity is not basically necessary if good angular resolution is available. Therefore, a continuous beam with very moderate energy selection as proposed by Wilkinson or a pulsed moderated reactor like Whitemore's, together with a high angular resolution and 4π detection system, may offer advantages in total intensity, assuming equal peak flux in each case.

In some examples it has seemed to me that people wanted to use neutrons where they could use x rays instead. If one has a splitting of a low order Debye-Scherrer line, then one can do this with x rays, unless it is only a magnetic line which would show such splitting. Therefore, I would not believe that it is a very important method except in the case when one can improve on general grounds, namely because the x-ray lines have a finite width; one can go beyond that limitation using neutrons, which I think one can do in some cases where back-scattering is present, or in using some of the focusing methods that have been mentioned.

As for conventional versus new methods, we are in a rather awkward position; we do not have the experience some of you have had, with the backing of 30 years of x rays and methods developed from that. We do not have a reactor yet to work on such problems and try them out. Nevertheless, some of us have to make a decision whether we should use pulsed methods or non-pulsed methods and I should like to say maybe it will be good for us if people like Wilkinson stay with the old methods and give us time--four years--to develop the new ones and then we will see who wins!

BURAS: I have found the splitting of the reflections (111) and $(\bar{1}11)$ for BiFeO_3 ; this is exactly the case of magnetic scattering and I would agree immediately one could do better with x rays if it is not a magnetic reflection.

In connection with the resolution and intensity discussion, we have just published a small report dealing with this problem, but I do not think it appropriate to discuss it here.

GLÄSER: I think it is worthwhile in connection with this discussion on different diffraction techniques to make just a short comment on a method we thought of recently. As shown in Fig. 8, this method uses a continuous white neutron beam impinging on a powder sample and neutrons are coherently scattered at a fixed angle. The neutrons then impinge on a rotating crystal or, better, a set of phased rotating crystals and are reflected whenever the Bragg condition is fulfilled. The different energies corresponding to different Bragg angles realized at different times are measured on a time scale. If one considers only one crystal, it looks just like the reverse of the continuous method. If one considers more carefully the properties of rotating crystals, one would find that by using a line of such phased crystals one can improve

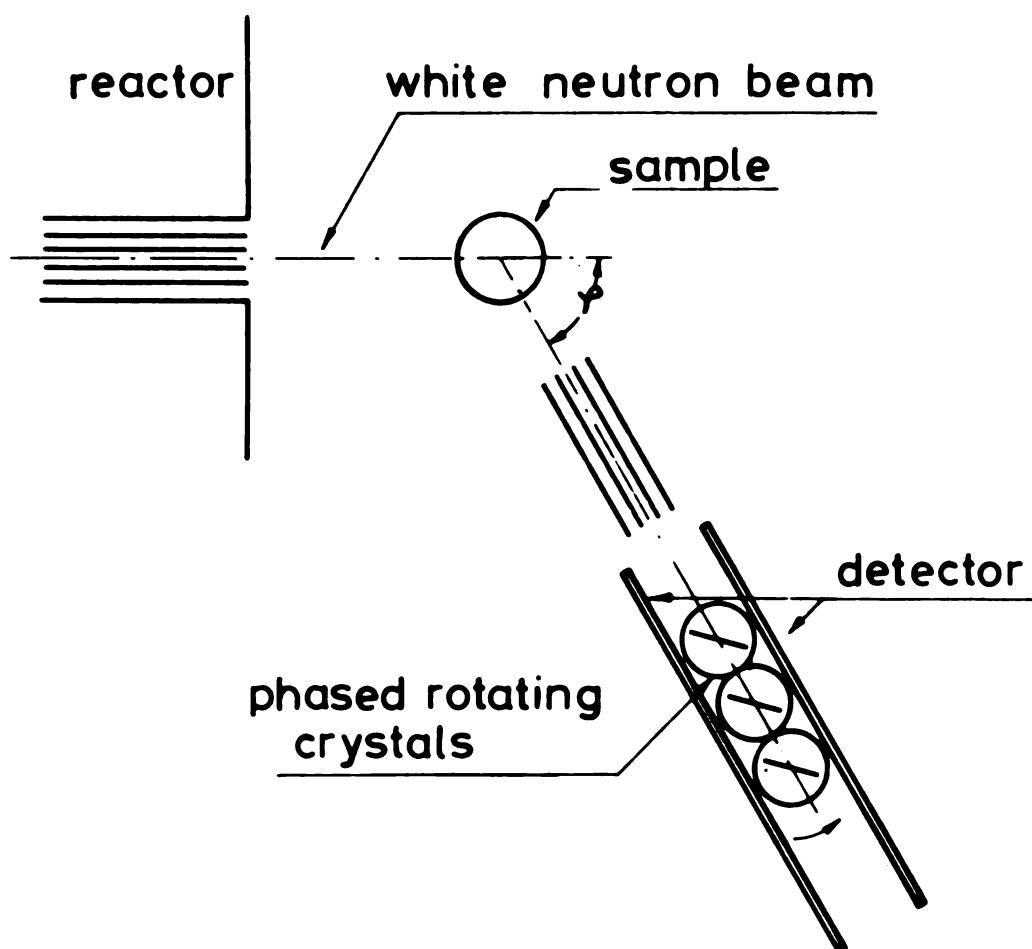


Fig. 8
ROTATING-CRYSTAL-DETECTOR DIFFRACTOMETER

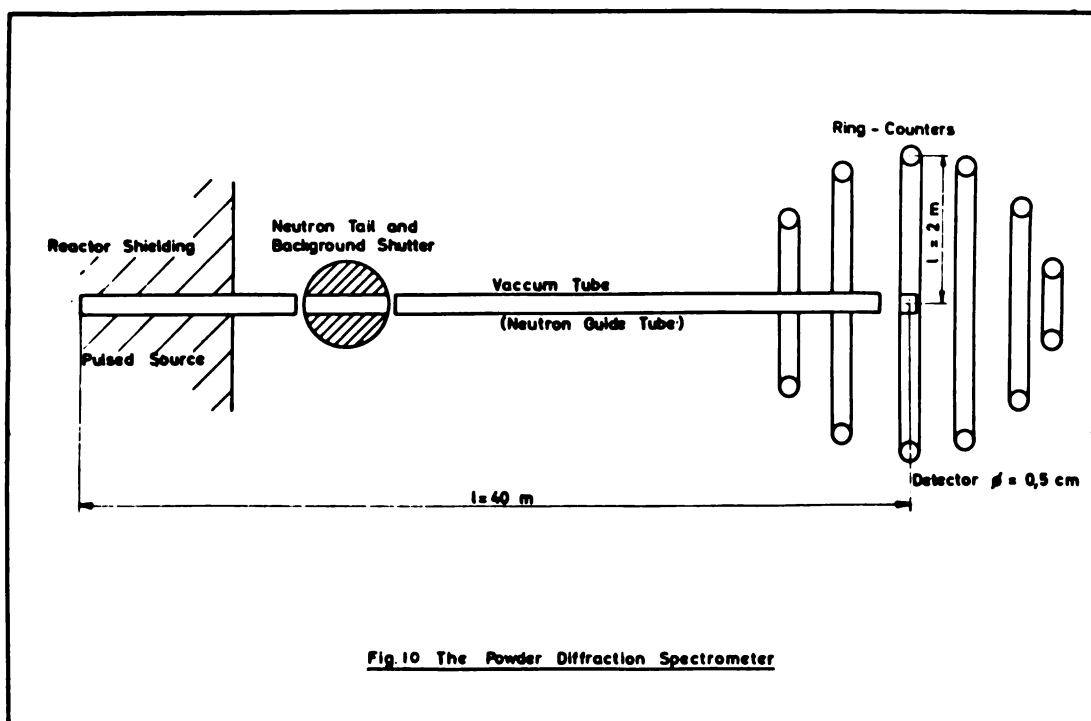
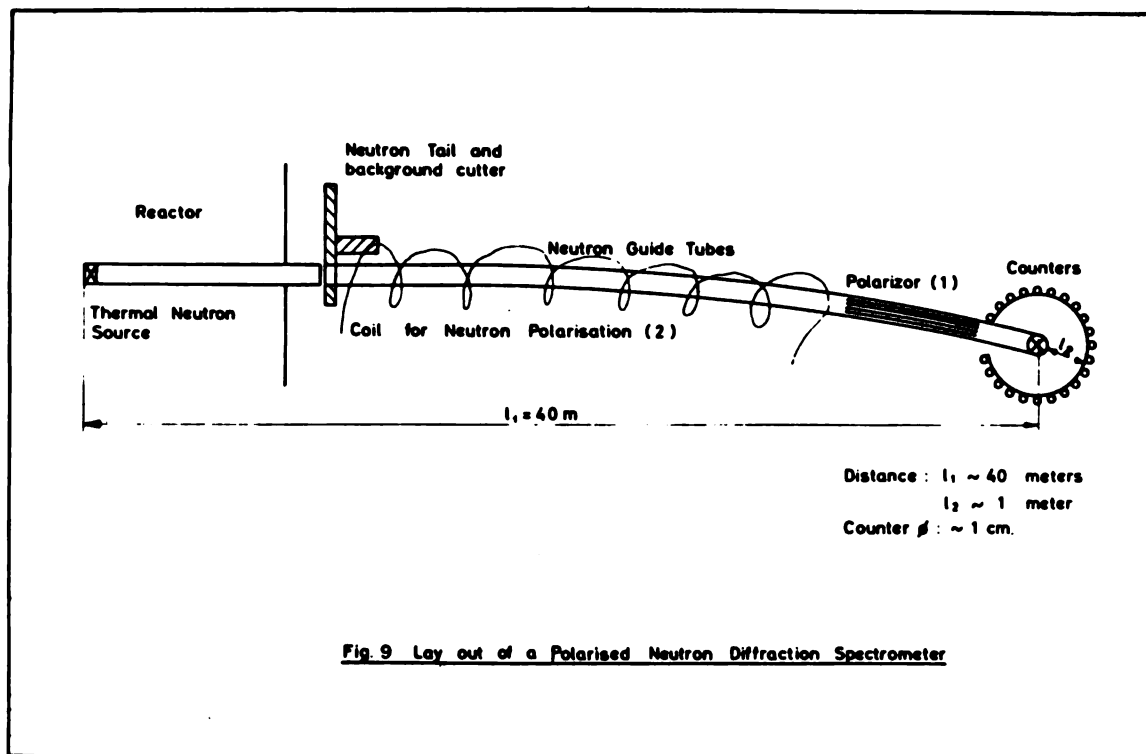
the intensity probably by a factor of 10 or so. With a preliminary setup using only one crystal, we demonstrated the feasibility of such a scheme a few months ago.

KLEY: I have a number of questions and I must also say that I disagree with a number of statements which have been made just now. The remark was made that it is impossible up to now to conceive of a TOF spectrometer with polarized neutrons. We thought about this problem some two and a half years ago; we thought of a spectrometer in which a neutron guide tube is used as shown in Fig. 9. A coil is wound around the guide tube and the inner surface is plated with nickel; while the neutrons are travelling down the tube, a current is put through the coil and when the neutrons are reflected from the surface they are polarized; since many reflections will occur, one can hope to obtain almost 100% polarized neutrons at the end of the tube. In case that this technique does not work as well as expected, we can use a bunch of magnetized mirror plates at the end of the tube to produce a beam of highly polarized neutrons. Therefore, I do think that this statement was wrong.

Secondly, the Panel indicated that there are problems in knowing the neutron spectra and intensity variations in a TOF spectrometer that do, of course, not exist in the conventional diffraction spectrometer which uses monoenergetic neutrons. Now I think it may be so in the TOF spectrometer arrangement of Brugger where the chopper is just in front of the sample and the long flight path section is between the sample and counters. A simultaneous measurement of the spectrum and intensity does indeed seem to be difficult in this facility. But if one uses a different setup, like the one of Buras, then I think one can monitor the shape of the spectrum with a thin $1/v$ counter during the measurement. The proper set-up would be: pulsed source, then a very long flight path, a $1/v$ detector just before the sample, a very short flight path from sample to detector, in order to profit from a large solid angle that can be obtained using ring counters as sketched in Fig. 10.

Then, there is the question concerning the second order. The Panel stated that there are no problems because it was proved at Brookhaven that by using germanium crystals, the second order contribution can be made very low or practically negligible. Now, it was remarked that for the study of organic crystals one needs long wavelength neutrons. In this case, the second order might not be negligible and mechanical devices will have to be used to suppress the second order and also produce a loss of intensity by a factor of 2 or 3. TOF spectrometers do not have the problem of second order.

Another question concerned the design of conventional spectrometers with multiple detectors. The Panel stated that in the conventional method one could use many detectors. Nobody has ever tried to use such a system; it has not even been tried at the new HFBR at Brookhaven. One of the difficulties with the installation of many detectors is the shielding of large volumes near the reactor surface. In the TOF method, this will be relatively easy because those detectors will be about 40 or 50 m



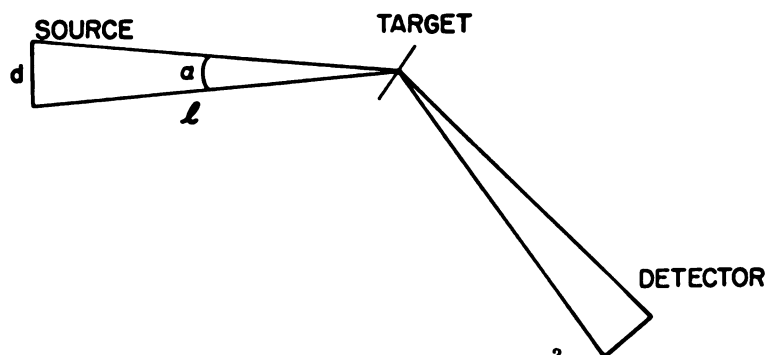
away from the reactor and there will be many counters, contrary to what Buras proposes. With ring counters one can measure in ten different directions at the same time as illustrated in Fig. 10. In this way one will use the source neutrons more efficiently.

BRUGGER: I have one comment about Kley's remarks. He mentioned that our method was not as good as Buras' for measuring the intensity. I think it is as good or better. We have a beam monitor in the beam at the same distance as our detectors; we not only see the flux that strikes the counter, but also how it is attenuated by the sample and attenuated by air.

As another comment it seems to me that the development of the TOF method is stimulating the steady-state people to rethink their problems and to consider possible means of improving their experiments.

KLEY: Caglioti made a statement on which I would like to have some more elucidation. That statement was that "neutrons are scattered from solids predominantly elastically, but there are very important exceptions." At the end of this discussion, I would like to have some information about these important exceptions.

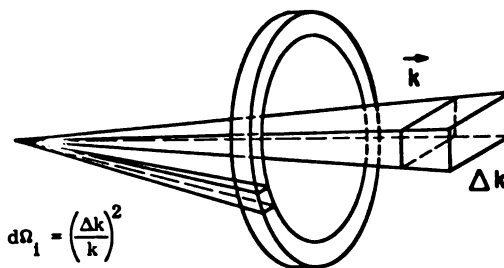
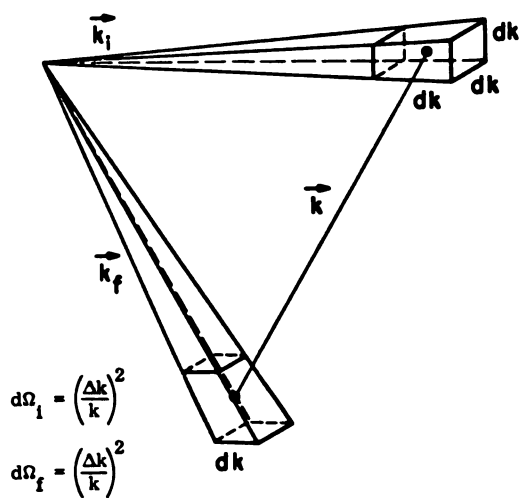
I would like to add a remark concerning the intensity of conventional and of TOF spectrometers. In Fig. 11, a sketch of the experimental situation in real and momentum space is given. On the very top of this figure one sees a scheme of the scattering experiment including the source, the target and the detector. Now in the classical conventional spectrometer, one has a neutron distribution of the scattered beam that is given by the collimator and detector size, the source and detector diameter are about of equal size. The experimental conditions in momentum space for the conventional and the TOF experiment are sketched in the lower left and right part of Fig. 11. This momentum space diagram technique used here has been introduced by Maier-Leibnitz (Nukleonik 8, Nr. 2, 61-67, 1966). It allows an excellent presentation of the resolution and intensity available in any scattering experiment. In particular it allows one to gain easy access to the optimum conditions for a given experiment. In Fig. 13, the collimation of the impinging and scattered beam is represented by the solid angles $d\Omega_1$ and $d\Omega_f$, respectively. In the conventional method, $d\Omega_1 \approx d\Omega_f = (\Delta k/k)^2$, as shown in the lower part of Fig. 11. The intensity at the detector, the important quantity, is proportional to the product of the initial and final solid angle. TOF experiments for powder samples can easily make use of ring counters. Therefore the final solid angle will be quite different than in the conventional spectrometer: it is $d\Omega_f = \Delta k \cdot 2\pi k \cdot \sin \alpha / k^2$. This solid angle is much larger than the initial one and gives a considerable increase in intensity: $d\Omega_f = d\Omega_1 \frac{2\pi \sin \alpha}{\Delta k/k}$. However, in TOF experiments we do have an important reduction factor, $f_R = \Delta t_s \nu_s$, the duty cycle of the pulsed source, that is of the order of 10^{-2} , and which is equal to unity in the conventional method using a continuous source. This reduction factor f_R (SORA) can be compensated by the ring counter design feature. Therefore, the intensity of the



$$y_{\text{Det}} = \left[\frac{\phi(E, t)}{4\pi} \times f_R \times dE_1 \times d\Omega_1 \times f_1 \times F_T \right] \cdot ND \cdot \frac{d^2\sigma}{d\Omega dE} \times d\Omega_f \times dF_f \cdot f_f \text{ [n/sec]}$$

$$f_R(\text{SORA}) = 0.25 \times 10^{-2}$$

$$f_R(\text{H.F.B.R.}) = 1$$



$$d\Omega_f = \frac{\Delta k \cdot 2\pi k \cdot \sin \alpha}{k^2}$$

$$d\Omega_f = d\Omega_1 \frac{2\pi \sin \alpha}{\Delta k/k}$$

$$\frac{y_{\text{T.O.F.}}}{y_{\text{static}}} = \Delta t_s \cdot \nu_s \frac{\alpha \pi \sin \alpha}{\Delta k/k} \cdot N_1 \cdot N_2 \cdot N_3 \frac{\phi_{\text{T.O.F.}}}{\phi_{\text{static}}}$$

N_1 = number of reflections

N_2 = number of measuring positions

N_3 = number of counters

FIG. 11. Geometrical and Momentum Space Factors for Neutron Scattering

TOF experiment should be higher by the factor $y_{\text{TOF}}/y_{\text{static}}$. The factor $\Delta t_s \cdot v_s$ is of the same order as $\Delta k/k$, since one works at least with a resolution in $|k_1|$ of 1% or better. N_1 stands for the number of possible reflections in one direction (10 or more). N_2 is a time factor that takes into account that in the conventional method one has to measure at least 10-20 different settings of angle in order to determine the width and position of the line, which information is obtained at once in the TOF spectrometer. N_3 stands for the number of ring counters that can be built. Assuming that $N_1 = N_2 = N_3 = 10$, we obtain a factor of 10^3 higher intensity in TOF than in the conventional method. Clearly, this "excess" in intensity could be used to improve the resolution of powder-diffraction experiments.

CAGLIOTTI: I think that it would be fair to compare experiments with only one counter because, if we consider several counters, then the thing gets complicated and, on the other hand, we could perhaps use more than one counter in connection with all kinds of sources. I see that there are several questions here. There is really not enough time for all of them, but let us hear Woods' comments.

WOODS: Brugger mentioned in his requirements for the pulsed reactor an upper limit of 50 pulses/sec. Would a faster repetition rate (say 200/sec) be a definite embarrassment in this diffraction experiment? For many inelastic experiments, the higher repetition rate would be advantageous.

BRUGGER: For some experiments, no, but Buras keeps mentioning the very large d-spacing planes which are obtained with cold neutrons and with a faster repetition rate we would get into an overlap problem. If we could suppress the size of the source and shorten the flight paths, we could go to a higher repetition rate. I pick my experimental values based on the other values I had in the table. If we perform inelastic scattering experiments in which we do not use the whole spectrum, then the overlap problem is not nearly so severe.

WOODS: Yes, that is why I asked the question. It would be nice to have a pulsed reactor which is suitable for both types of experiments. The higher repetition rate would certainly be an advantage for many experiments anyway.

BRUGGER: Yes, a higher repetition rate for some experiments would be a definite advantage.

WILKINSON: I missed one point on Kley's comments about polarized neutrons. I was not sure whether this method of polarization had been built and was actually tested or whether this was something that he believes will work.

KLEY: We have built one iron-cobalt mirror at Ispra, which gives 90% polarization. But we have not yet built the tube so we do not have this device, but it is known from mirror reflections that one can probably get as high a degree of polarization as with the crystal.

WILKINSON: And what intensity do you get?

KLEY: If the required angular resolution is within twice the total reflection angle $\theta = \frac{0.01}{k}$ (k in \AA^{-1}) then we can use neutron guide tubes and not lose much intensity since very high values in reflectivity have been measured, at least for glass plates. If I have understood your comments correctly, then one very often needs an angular resolution of the order of 10 minutes of arc. Now this is within the possibilities of neutron guide tubes and therefore full intensity can be obtained at the end of the beam tube, since the intensity is determined by the initial collimation angle $d\Omega_1$, which is guaranteed by the tube. This consideration supposes that the TOF experiment is carried out with the same energy resolution as the conventional method. Really, only the neutron guide tube technique [H. Maier-Leibnitz and T. Springer, Reactor Sci. Tech. 17, 217 (1963)] allows one to get the necessary energy resolution with pulsed reactors without appreciable loss in intensity.

WILKINSON: As I remember, in some of the early experiments on magnetic mirrors for obtaining polarized neutron beams, the intensities were quite small.

KLEY: As a matter of fact, for one single mirror of 1-m length and 10-cm height, 10^7 n/sec are obtained at the CP5 reactor at Argonne. I think it is the highest intensity which was ever achieved.

BURAS: When I mentioned one detector, I referred to single crystal measurements.

CAGLIOTI: I should answer a remark Kley has addressed to me. The question was about the amount of inelastic contributions in a conventional diffraction pattern of a solid. I have some figures prepared for this matter. Figure 12 shows a conventional diffraction pattern of NH_4Br taken with a two-axis spectrometer (G. Caglioti and F. Pompa, Nuovo Cimento B, 1966, to be published). In this diffraction pattern one detects contributions from all of the scattered wavelengths arising from the impinging monochromatic neutron beam; there is some inelastic scattering because hydrogen has a high incoherent cross section (80 b) which is associated with mostly inelastic scattering processes. This high incoherent inelastic cross section is mainly responsible for the enormous background underlying the Bragg reflections in Fig. 12. A way to avoid part of this background, especially when dealing with hydrogen-containing compounds, is offered by the fact that the signal, i.e., the Bragg reflection, is correlated with processes which are essentially elastic. An elastic diffraction pattern can be obtained by using a three-axis spectrometer and setting the analyzing spectrometer so as to accept only neutrons in the detector which, within the resolution, did not suffer any energy transfer as a result of the scattering process. Figure 13 (G. Caglioti and F. Pompa, loc. cit.) shows this result, namely the elastic diffraction pattern from the same non-deuterated powder specimen of NH_4Br . One sees that the enhancement is really formidable in the signal-to-background ratio.

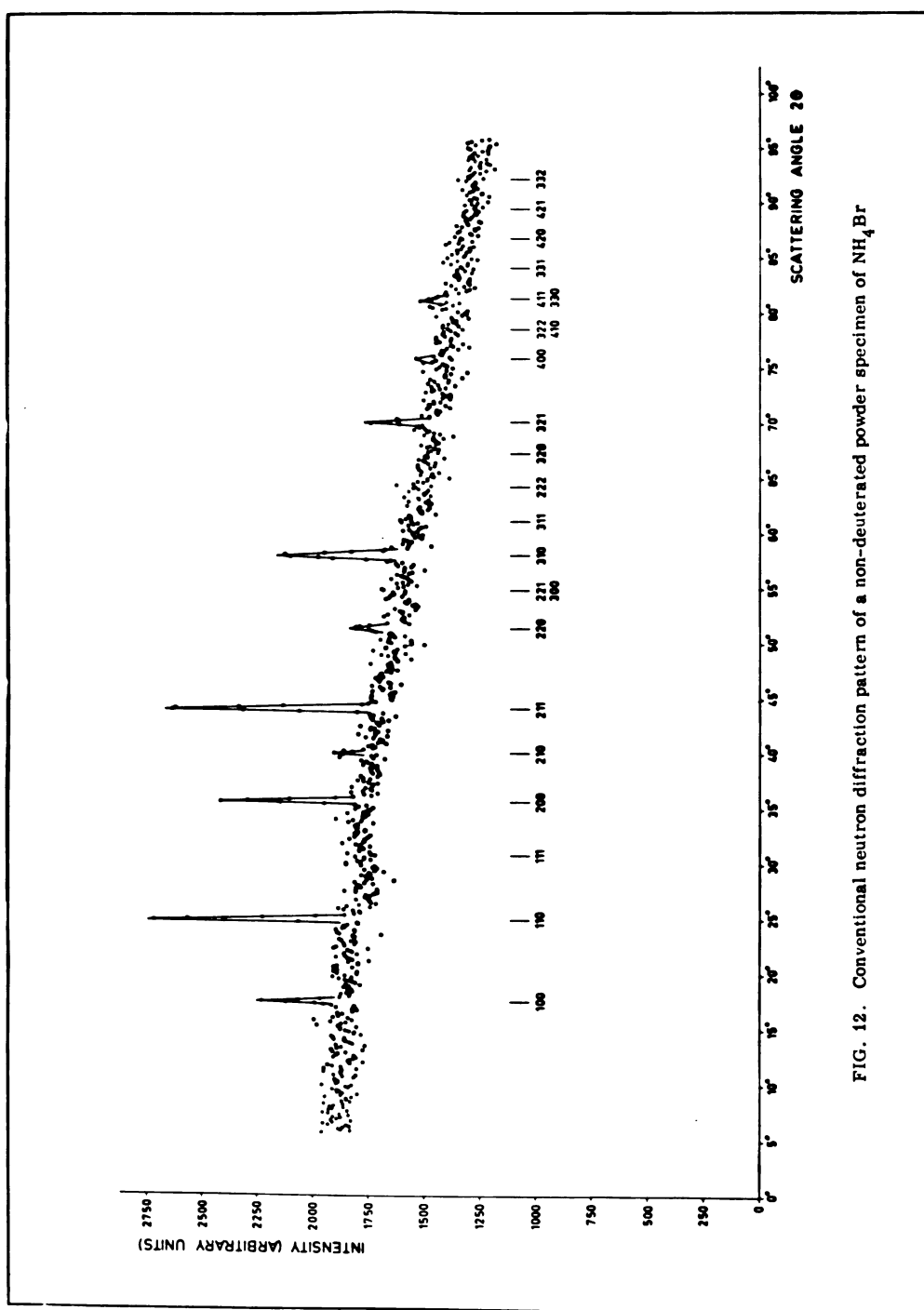


FIG. 12. Conventional neutron diffraction pattern of a non-deuterated powder specimen of NH_4Br

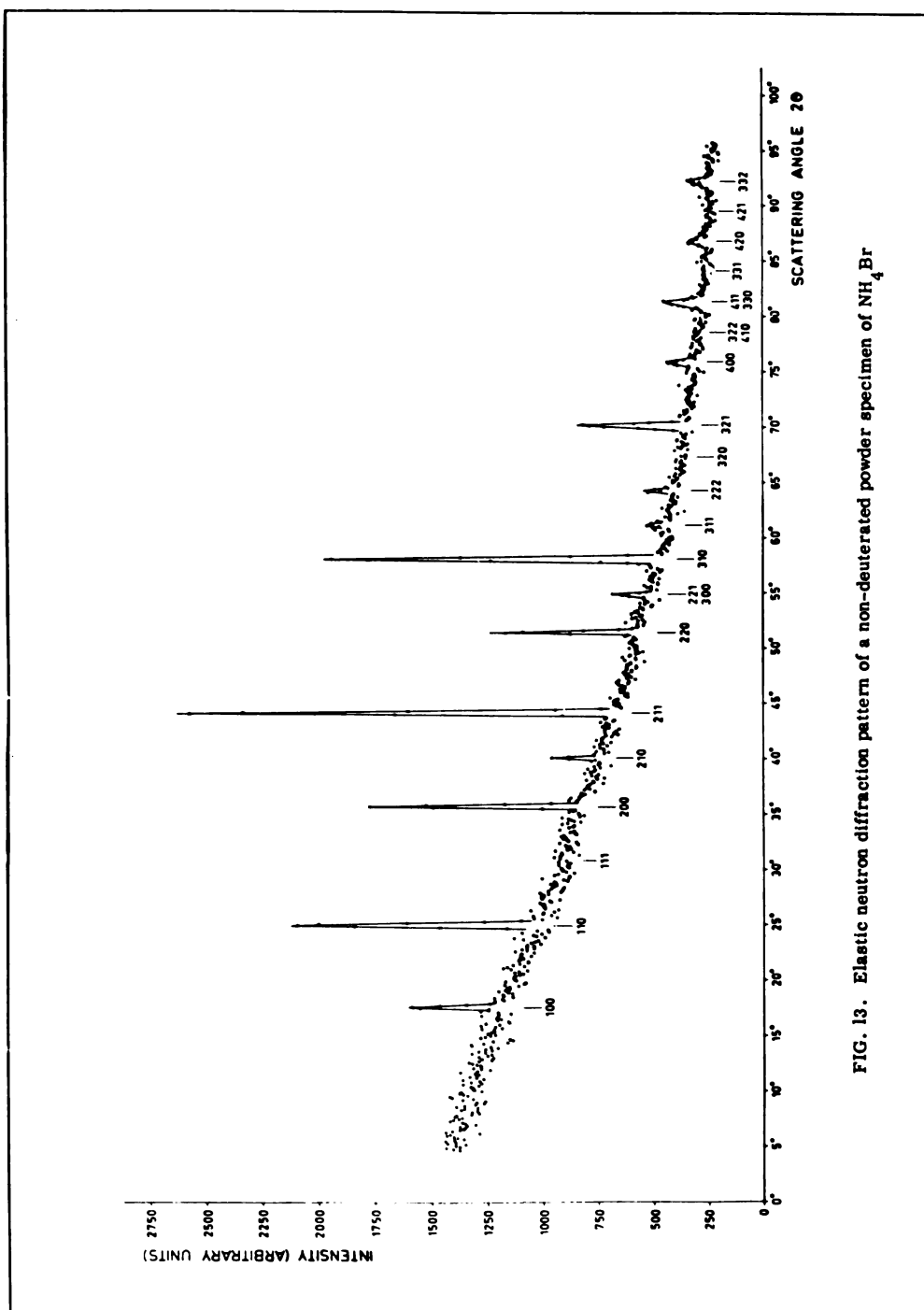


FIG. 13. Elastic neutron diffraction pattern of a non-deuterated powder specimen of NH_4Br

This method of elastic neutron diffraction, if utilized in connection with pulsed neutron sources, would eliminate some of the points which Wilkinson stated against the TOF method; for instance, in this case, extinction corrections will not have to be applied to several wavelengths and the calibration of the incident neutron beam would be straightforward because it would always refer to the same wavelength. This method could be used effectively in the case, outlined by Wilkinson, of the study of static-type disorder scattering. Of course use of the above method of elastic diffraction would imply a reduction of the intensity, but perhaps if sources of higher intensity are available this method could prove of value there.

WILKINSON: I would like to just make one comment in answer to the previous remark, because it seems to me that your technique is actually another technique of obtaining monochromatic radiation. I do not consider this the conventional TOF technique.

CAGLIOTTI: Well, this I think is a TOF technique. When used in connection with pulsed sources, this method would just imply an analyzing spectrometer pivoting around the sample, set for a given wavelength. The TOF between the source and the sample could then be measured for any detected neutron.

WILKINSON: That is my point; you are using neutrons of one wavelength so that it is not the "conventional" TOF technique.

CAGLIOTTI: That is right--it is another technique which takes advantage of the extra intensity in the source, which we are always ready to utilize when we are supplied with it.

BEYSTER: I would like to comment briefly on the specifications for a pulsed source which Brugger mentioned. Some of these specifications are very similar to those we have been discussing here, and I thought I would point out where they are similar and where they are not. For example, a 15- μ sec pulse is the sort of pulse one can get from an accelerator-booster plus a highly multiplying assembly. The matter of how one makes the source is not as yet completely certain. There are at least two possible ways that have been investigated. One is to poison the moderator; this seems to work from the evidence we have so far in shortening the pulse without losing peak intensity. The other scheme which has been studied more thoroughly in England by Poole, and discussed previously, consists of departmentalizing the source. Actually, detailed calculations have been made on the burst shape in that case and one gets, for a 1-MW average power booster, a flux of about 10^{15} peak and a pulse width of 15 μ sec in that geometry. So it looks as though with a presently feasible booster (i.e., one that is not too much of a strain on the present state of the art, in other words, a booster which could be built on an accelerator that has been built or will be built in the next year or two, namely one with ~ 50 kW average power) that one can get the pulse shape and intensity desired within a factor of 10, not within a factor of

100. In other words, we are not low by a factor of 100 as Brugger may have implied.

The other question which comes up is the matter of repetition rate because if one cannot take advantage of the high repetition rate available from the booster then one loses in figure of merit or in time-averaged flux on the sample. In the calculations which I have done comparing the mechanically pulsed reactor with a burst of around 75 μ sec with this 15 μ sec wide burst with an accelerator-pulsed reactor, one can gain a factor of 5 in intensity or flux on the scattering sample if he can take advantage of higher repetition rates of, say, about 250 pulses/sec. Now it looks to me as though for the short flight paths that one would use in this kind of an experiment, namely, less than 5 m, that this is possible without using any auxiliary rotating equipment. I did want to ask Brugger if it is not possible to stop the overlap, i.e., is it possible to use auxiliary equipment as rotating collimators to eliminate the overlap problem.

CAGLIOTTI: I am very sorry to have to stop the discussion at this point. I would like just to summarize very briefly, I think, as Maier-Leibnitz told us, that some time will be required to develop properly the TOF method. And, in the course of a few years, most of the questions which came about today will no doubt be duly answered. In any case, I think that experimentalists would like to have more and more intense sources, and that they would always be ready to utilize them. One thing seems obvious to me, which was already stated in the July Panel at Dubna on repetitively pulsed neutron sources; if we compare sources their cost increases roughly with the average power; what we are interested in is the peak power. If we are going to produce good technical means for utilizing TOF methods, we should really try to build at least one source which should be at least an order of magnitude more intense than the one already in existence at Dubna.

SESSION VI PART B - PANEL DISCUSSION

ON

INELASTIC SCATTERING

Panel Members: H. Palevsky, Chairman
W. Gläser
B. Jacrot
W. Kley
A. Woods

Secretary: J. Yarnell

PALEVSKY: When our Panel met to discuss how best to serve the purpose of this meeting, we found ourselves spending a good part of the time trying to define this purpose, and, since the Panel was composed of experimenters, it was clear to me that a lively discussion would ensue, each of us supporting the merits of our own particular technique. I urged the Panel to try before we convened for this meeting to consider if there was some common ground that could serve as a guide to reactor designers and satisfy the need of all users in our field. We were able to agree that there were indeed a few invariants and, as Chairman, I was delegated to preface this Panel Discussion with a few words about these common requirements. First of all, we agreed that the question of which is the best neutron source and measurement technique could not be answered in the general context. One must carefully define the particular experiment that is being contemplated before a truly meaningful answer can be deduced. There are many factors that go into each experiment, and the conclusion as to the best approach many times depends on the assumed boundary conditions.

Now, what we could agree on is the following. For a broad class of inelastic scattering measurements comparable results can be obtained with various combinations of reactors and associated equipment. By comparable results we mean a certain counting rate for a given resolution. What determines the magnitude of these parameters is primarily the peak flux at the radiating surface of the reactor and the available solid angle. Now with a little bit of thought you conclude that the preceding statement cannot really be correct. That is indeed true because on the basis of this statement one would say immediately that you just build a device that gives you a very high peak flux and a very low repetition rate. What I have neglected to say is that, on the basis of what we had heard earlier this week, we have assumed what we think are reasonable duty cycles for pulsed systems of the order of 10^{-2} or 10^{-3} . Now it turns out that under these conditions the diffraction and time-of-flight (TOF) techniques give about the same results. The integrated intensity at one wave length of a diffractometer plus steady-state reactor is balanced by the accumulation of data at

many wave lengths by the time-of-flight devices. Therefore, I think it is important to reiterate to reactor designers to take with a grain of salt some of the very strong statements that some of our participants will make and really to understand that what we need is high flux and large solid angles.

I will first call on Woods to present his comments.

WOODS: There are a few points which I would like to make which arose out of the discussion of Brugger's paper and which have come up since. We are actually at a disadvantage when trying to justify big expenditures on high flux reactors, compared with people in high energy physics, for example, where they cross definite thresholds and can propose definite new qualitative experiments which cannot possibly be done with existing equipment. We are not in a position to do this; all we are getting is higher intensities and all we are going to be able to do with them is to do somewhat better the things that, in principle at least, we can already do now.

What these increased fluxes will do for us can be broadly put into two classes. We can do the experiments which are at the moment difficult in a reasonable amount of time just because of the length of time it now takes to collect the data. In this class of experiments are those in which one needs to measure scattered neutron intensities.

The other class of experiments which we can do much better if we have big increases in neutron flux are those which require improvement in both energy and momentum resolution. Brugger mentioned both classes of experiments in his paper and I just want to add a little bit of weight to what he said. I think it is fair to say that very few inelastic neutron scattering experiments which require a knowledge of the intensity of the inelastically scattered neutrons have been done in a completely satisfactory manner up to the present time. Many experiments have been attempted to measure the intensity of inelastically scattered neutrons, but few, if any, of these have been done with the high precision and completeness that is required to extract really good physics from them. If we had a factor of 10 to 100 increase in available fluxes, there is no doubt that we could make a tremendous improvement in this state of affairs. This was illustrated in Brugger's talk by the case of vanadium where at least a half-dozen different groups since the mid 1950's have attempted to measure the frequency distribution function for vanadium and he showed that there was really quite shocking disagreement between the various experiments. Not that I necessarily believe that knowing the phonon frequency distribution function for vanadium is one of the outstanding problems facing us in solid-state physics, but it does serve as an example of one of the simpler experiments where it is necessary to measure inelastically scattered neutron intensities and we just have not been able to do a good job of it as yet. There are many other cases where it would be very desirable to measure the complete scattering function, $S(q, \omega)$, over a large range of momentum transfer and energy transfer and much higher fluxes will be necessary before we really feel

satisfied that we can do an adequate job of this sort of measurement. The other area is in those experiments which require high resolution. While it is true to say that if you get a factor of 10 to a 100 increase in neutron flux that you can only improve the resolution of your apparatus for this sort of experiment by something like a factor of 2, this does not necessarily mean that we are in a losing battle. For many experiments the overall quality of the experimental results is not proportional to the resolution of your equipment but it goes up as a much higher power than this and is, in my opinion, much more nearly proportional to the increased flux which is available. Such a statement of course depends on the particular experiment that you are thinking about, but it is not hard to think of an example where a very modest improvement in experimental resolution would be very helpful in interpreting the measurements which you make.

To the pessimists who say that this factor of 2 will not really be a particularly useful improvement and that this factor of 10 to 100 would hardly be worth having because you can only improve the resolution by a factor of 2, I would ask them to go back and worsen their resolution by a factor of 2, then be content to do their experiments with reactors of 10^{12} or 10^{13} n/cm²/sec. I think this would, in fact, make them rather unhappy.

Now to answer the questions which were asked about what sort of flux we would desire, the answer is a very simple one. We want the highest flux that we can possibly get and, if possible, we would like to have this flux in the steady state, which brings me to the next point. As most of you know, there are two generally accepted ways of doing inelastic neutron scattering experiments. There is the pulsed beam, the TOF technique, and the crystal spectrometer technique. The arguments in favor of one or the other of these two approaches have been given many times in the past and are well-known to the people who are actively engaged in the field, and I do not have time to go into the detailed operation of each of these two methods of doing experiments. It suffices to say that when you are doing a TOF experiment, given a steady-state source, you must throw away something like 99% of the beam of the wave length which you want. (I am assuming here that you do want a mono-energetic incident neutron beam. This is not to be confused with diffraction experiments we have heard about this week, many of which can be done using the whole spectrum from the reactor.) Now the TOF method can readily allow one to get a lot of this back; it is quite a simple matter to put a large number of detectors around the specimen and gain a further advantage in these measurements--one can look at all scattered neutron energies simultaneously so one is able to get back most of the intensity that had been thrown away in order to pulse the beam in the first place. The crystal spectrometers use all of the incident beam, except what they lose by reflectivities from the various crystals, but that is rather a relatively minor thing for the purpose of the argument, at least. However, in the scattered neutron distribution, it is necessary

to determine the energy of the neutron after it has been scattered from the sample, and in general the detector can be set to look at only one scattered neutron energy at a time. If you want to observe a distribution over a large energy range, you have to set up your detector to look at various energies one after the other so that the experiment takes a long time. It is possible to put in more than one detector but it is certainly more awkward than it is with a TOF spectrometer. With present detectors, it is only possible to look at one energy at a time. Thus, the TOF spectrometers, because they can look at all energies and a large number of scattering angles at any one time, are most suited to those experiments where you want to measure the complete scattering function, $S(q, \omega)$, which was introduced in the previous Panel. The crystal spectrometer, on the other hand, is best suited to those experiments where you want to measure particular levels of the system which would have one wave vector, i.e., one momentum transfer associated with them and a single energy. You cannot look at just one energy, you have to look at a range to see where your peak is, but if you are just interested in a very sharp peak in the scattered distribution, you only have to look at, say 15 different points with your analyzing spectrometer. If you had information over the rest of the energy spectrum, you would just throw it away. The particular advantage of the crystal spectrometer for this sort of measurement is that it is very readily programmed; the angles can be varied in a very straightforward and simple manner so that you can choose the momentum transfer at which you want to take an energy distribution before you do the experiment. Then you can pick a very small energy range and just look at that so you finally collect very little data perhaps in a given period of time, but you do specifically the experiment that you want to do. In the TOF spectrometer this sort of control has not been accomplished in any sort of satisfactory manner. (I am not saying it is impossible, but it is certainly much more difficult to do.) We have an analogy that we like to draw at Chalk River and that we have used several times in trying to explain the difference between TOF instruments and crystal spectrometers for measurements of lattice vibrations in single crystals, which is one of the areas in which we have been most interested. Consider a man hunting ducks. There is a flock of ducks flying overhead and he has two guns with him, a shotgun and a high-powered rifle with a telescopic sight. Now the TOF spectrometer is like the shotgun--all he has to do is take the shotgun and shoot into the flock of ducks (he does not have to take any particular aim); the chances are he will get a large number of ducks. Some of them may be pretty thin with not much meat on them but, on the other hand, he can go home and say, "Look at all the ducks I shot today." Whereas, by looking through the telescopic sight on the high-powered rifle, he sees one particularly fat one that is obviously the choice of the whole flock and knows that, if he can shoot down that particular duck, he will certainly have enough to feed him well on that particular day. This is more similar to the situation with the crystal spectrometer; you can pick out the phonon that you want

to study--it may be a very crucial one in the analysis of the problem that you are interested in. There are many examples where the behavior of certain particular phonons has, in fact, contained a large amount of the information which you would really like to know about the system.

Now, I am sure that Kley is going to tell you about the beautiful advantages of the TOF apparatus later on and I will say right now that I am prepared to agree with almost everything he says. There is one thing about it that I think is important to bring out and that is, when you are doing an experiment with the crystal spectrometer to look at a particular phonon, you can optimize your spectrometer for resolution and intensity so that you can get the best result for that particular one. If you use the TOF method, you may collect ten times as much data in a given amount of time, but it is obviously very difficult, if not impossible, to choose your resolution so that it is optimum for all of these phonons simultaneously. Naturally, if the maximum flux which we want in our reactor could be put into short time bursts so that it would be pulsed, it would be unforgivable if we did not make use of TOF techniques in spite of the obvious increase in cost and complexity. I am sure there would be tremendous advantages in going to TOF techniques if we could take our steady-state 10^{16} flux and put it into 5- or 10- μ sec bursts, so that we have an effective flux which is very much higher.

KLEY: I think the comparison Woods made with the precision gun and the shotgun is really correct. It is the old comparison: what do you like, quality or quantity? Now, we want to have both, of course, and I think he said it very clearly; with the crystal spectrometer, you get quality. But I think we can probably arrange it that we can have both quantity and quality in the time-of-flight spectrometer. Now to do this, I have to say a few things about the burst width conditions for thermal and cold neutron experiments because in the last few days and particularly in Panel A, there have been quite a number of statements made that you need very short pulses, and much shorter pulses than the devices produce nowadays. Now if I understand these statements correctly, this means one would like to operate choppers with much shorter burst times and pulsed reactors deliver bursts that are too long. Now nobody took the care to justify these statements. I think they have been based more on authority than on analysis. Therefore, I will try now to outline what are really the conditions and see if one cannot find some conditions to define the kind of burst widths you need in TOF experiments.

(The following is a written summary of his remarks, prepared by Dr. Kley.)

BURST WIDTH CONDITIONS FOR THERMAL AND
COLD NEUTRON EXPERIMENTS

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Department of Reactor Physics
EURATOM C.C.R. - Ispra

ABSTRACT: It is shown that for a large number of experiments, including inelastic and elastic scattering experiments, where not only the energy, but also the momentum transfer, has to be measured accurately, the burst width is determined by the diameter of the neutron source and the velocity of the neutrons used in the experiments: $\Delta t = d/v$, d = source diameter and v = neutron velocity. (i.e., $\Delta t = \frac{0.1 \text{ m}}{2200 \text{ m/sec}} = 45 \text{ } \mu\text{sec.}$) In analogy to Bragg reflected neutron beams this burst width condition is comparable with matching the collimation and the mosaic spread of the Bragg crystal. If thermal and cold neutrons are being used from a pulsed neutron source, then one can have two different situations:

1. The source is being kept at very low temperatures, so that the neutrons of interest have an energy $E_n > kT$, where T is the moderator temperature, or
2. one adjusts the moderator temperature T so that $E_n \approx kT$.

In both cases, the burst width of the neutrons of interest will be approximately $\Delta t_s \approx \frac{1}{v}$. There is a certain coherence between the experimental requirement ($\Delta t = \frac{d}{v}$) and the thermal and cold neutron pulse characteristic of a pulsed reactor.

I. Introduction

Choppers at steady-state reactors can produce in principle any burst width that is required for an experiment. This capability made choppers an extremely versatile tool for many different types of experiments. Some of these advantages have to be given up if one uses pulsed fast neutron systems with associated thermal and cold neutron moderators. In order to obtain the highest thermal peak flux, the moderator has to have a certain minimum size.

Figure 1 shows for what thickness one obtains saturation of the thermal flux in a H_2O -moderator. The saturation thickness is about 7 cm. There exists in general an optimum condition for the geometrical shape and material composition. Figure 2 and Fig. 3 show results that have been obtained (1) by pulsed neutron measurements using a 1 MEV Van de Graaff

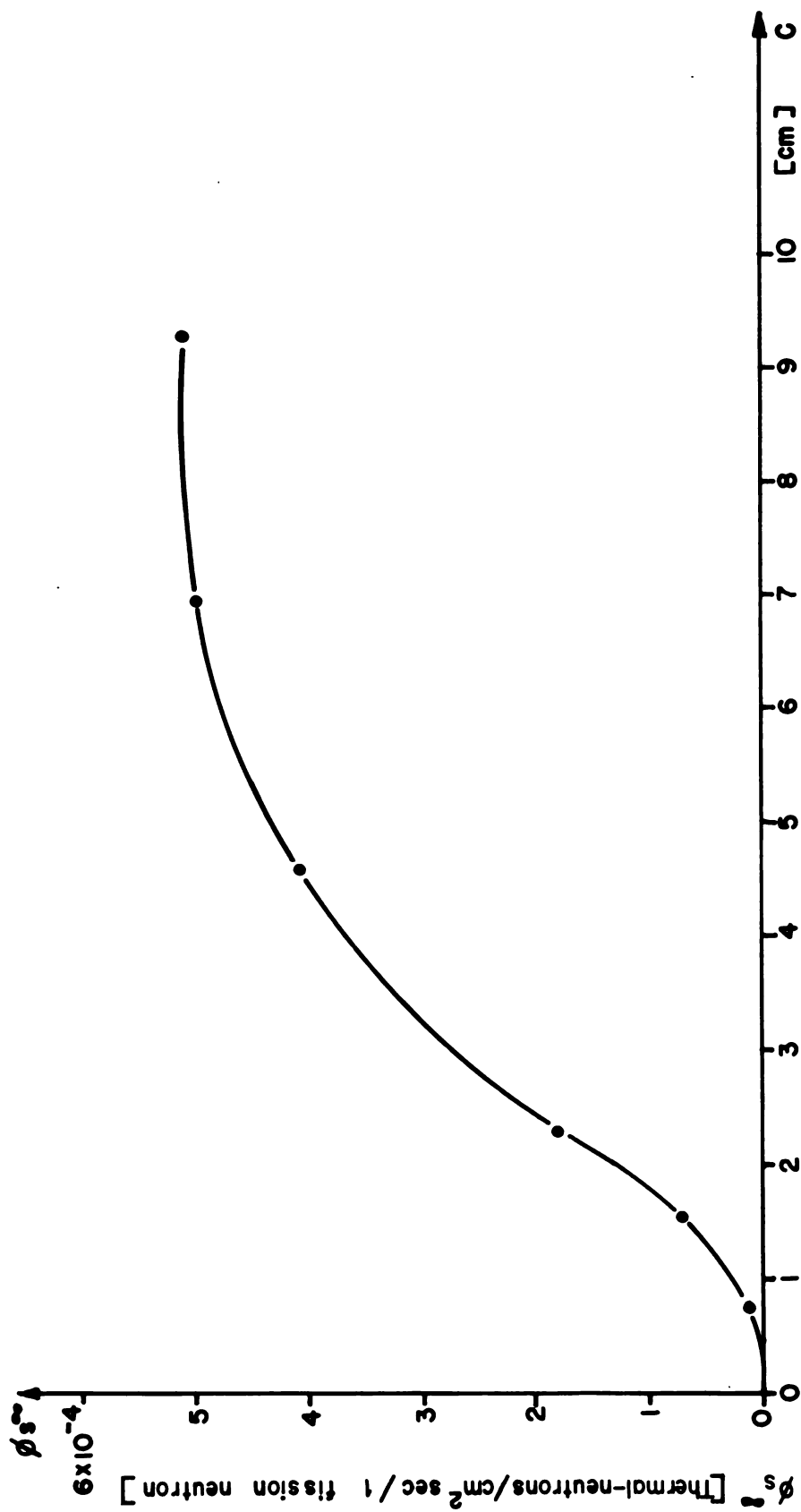
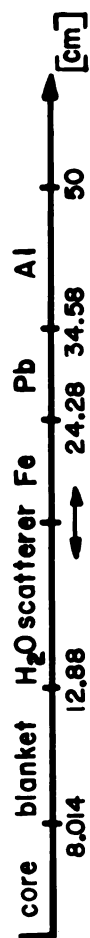


Fig. 1. The maximum static thermal flux ϕ_s^∞ in the SORA-H₂O moderator as function of the moderator thickness. For the one dimensional Sn-calculation the following geometry and material composition was used



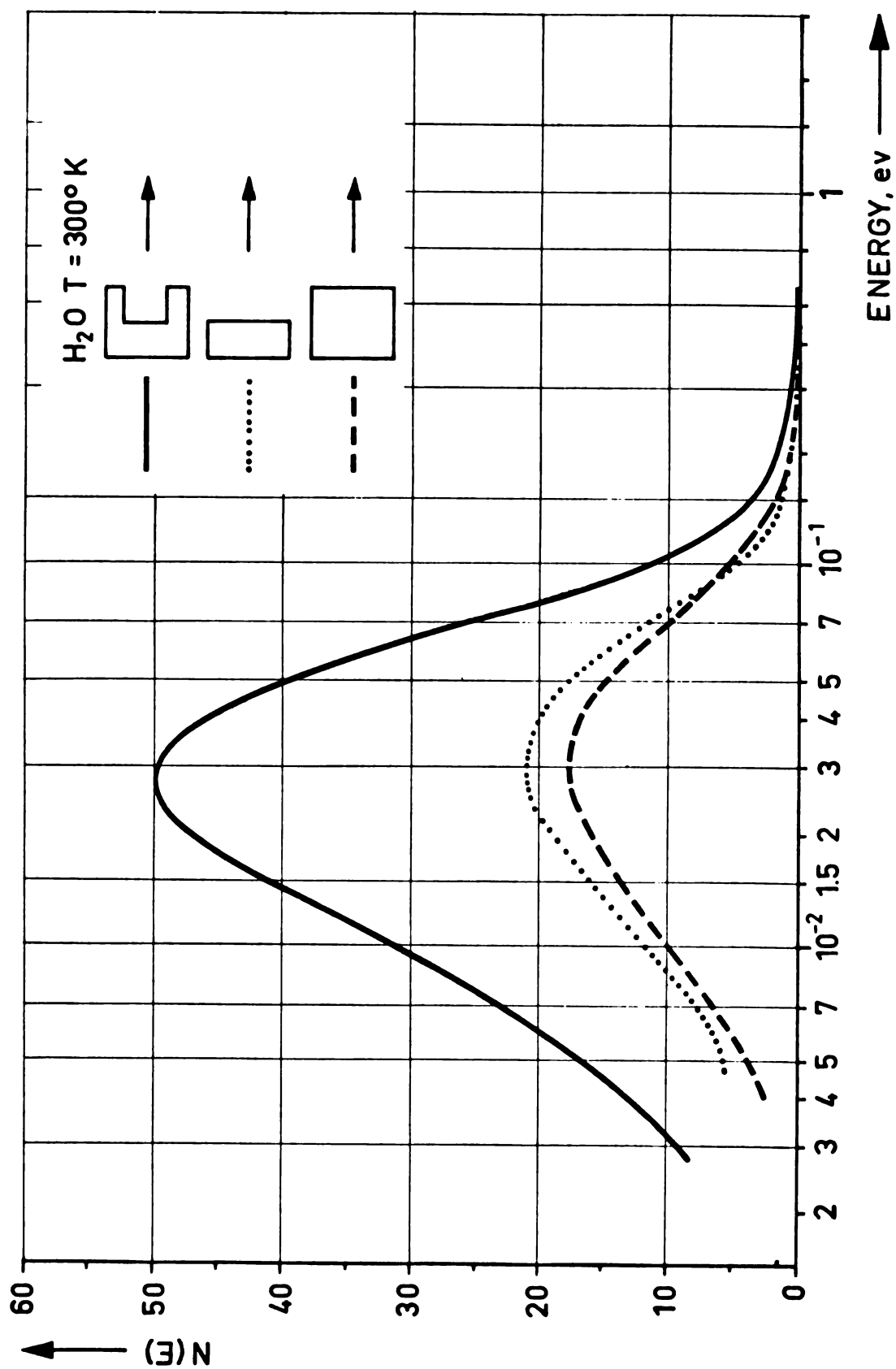


FIG. 2

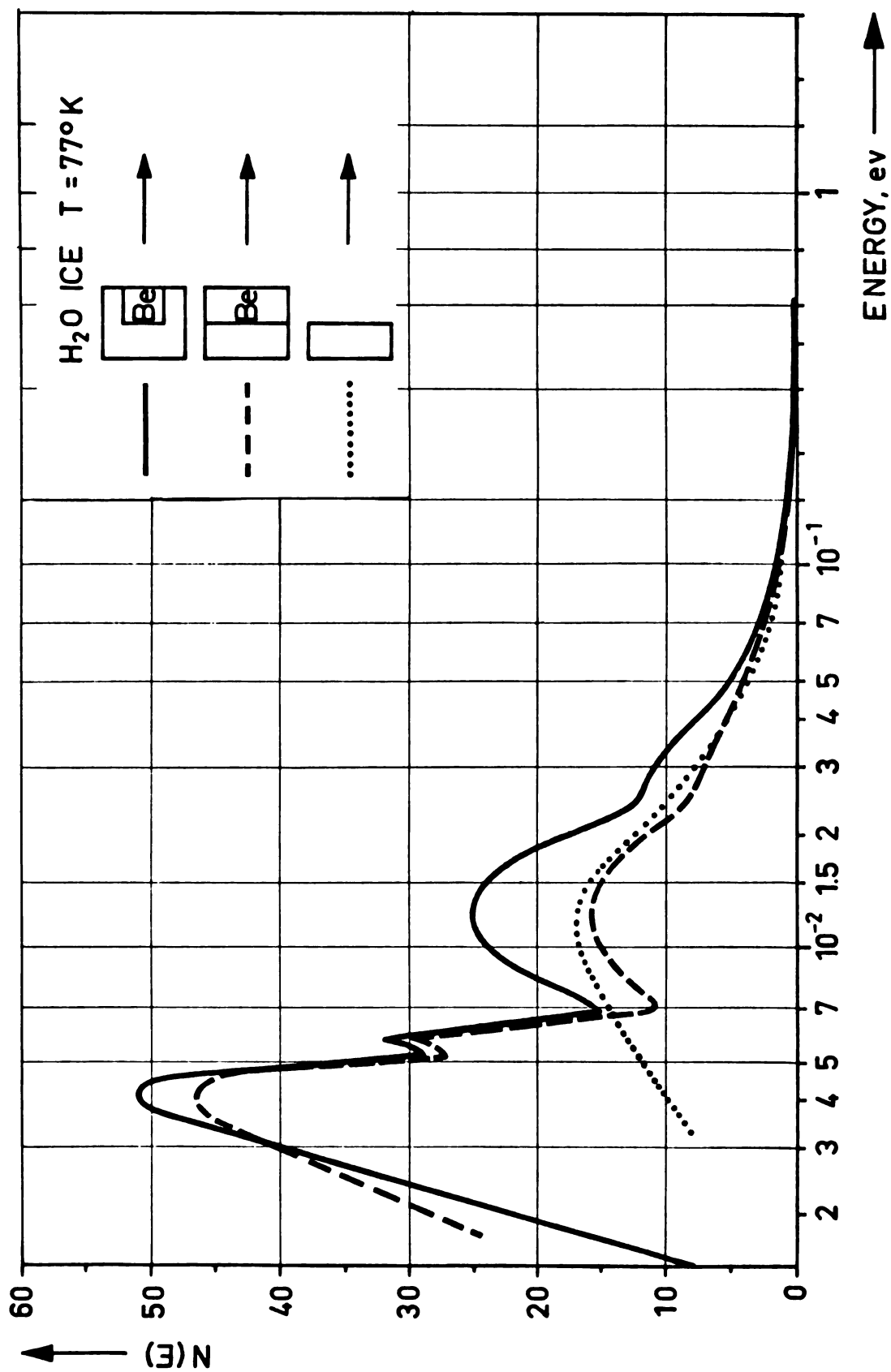


FIG. 3

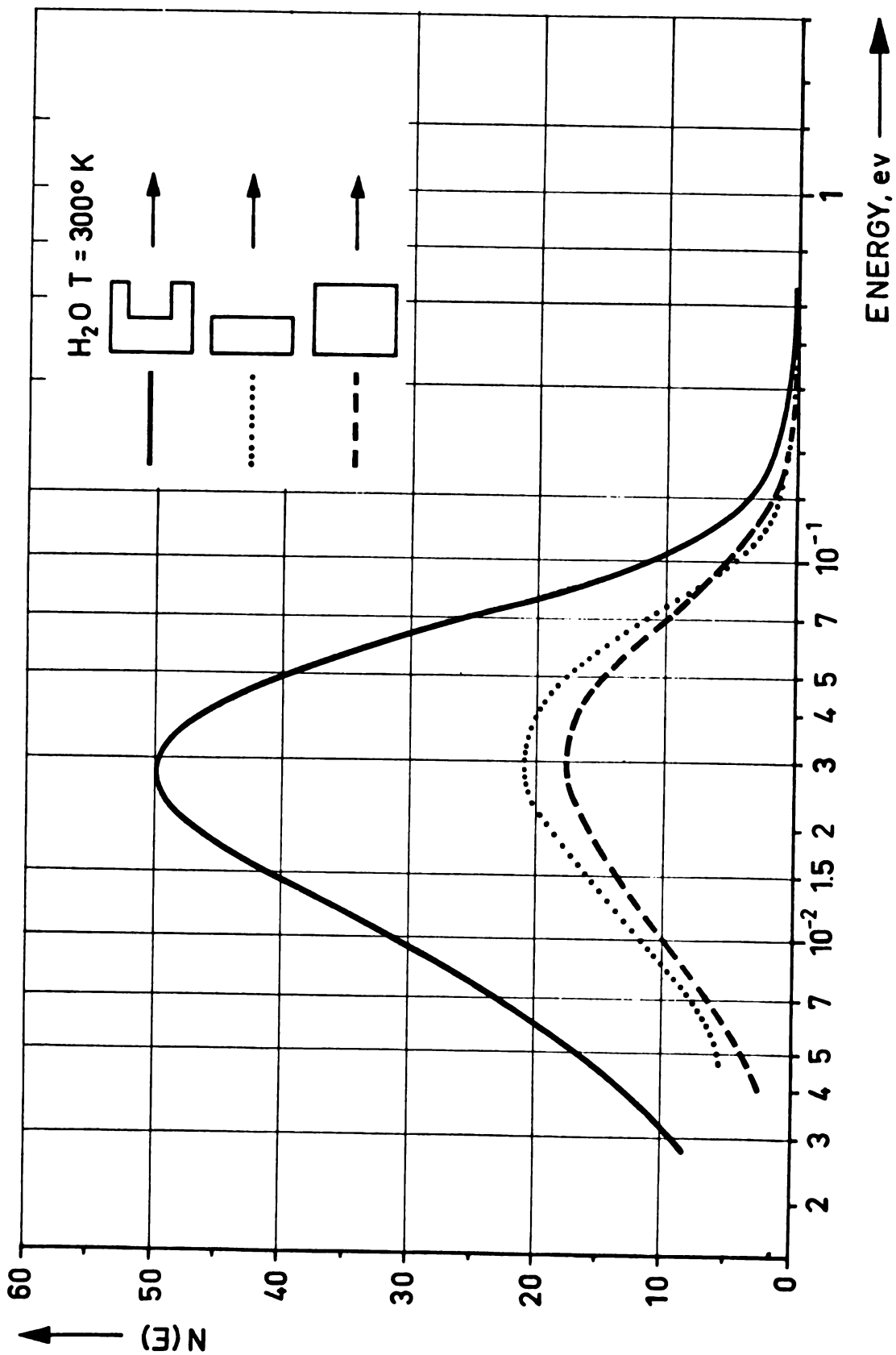


FIG. 2

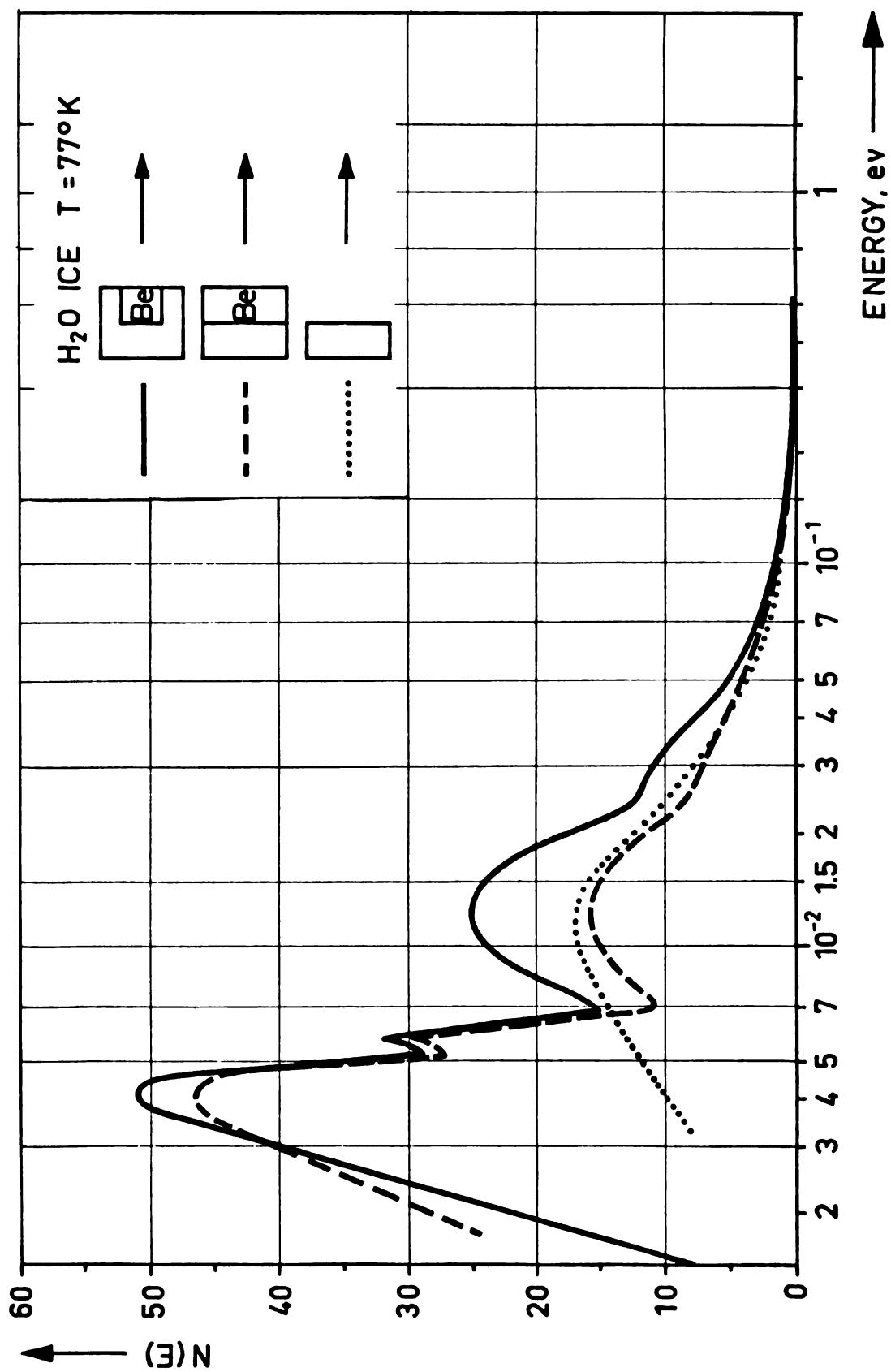
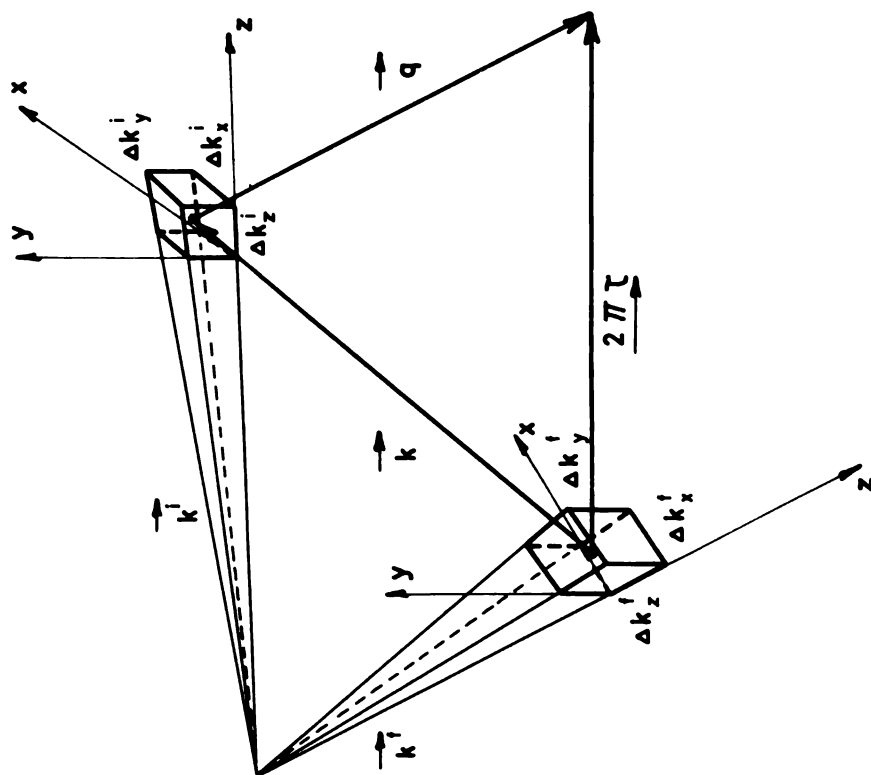


FIG. 3

and a cylindrical moderator cryostat with an inner diameter of 11.6 cm and an overall height of 15.3 cm. For such a pulsed system, the thermal neutron burst width is, however, energy dependent, and varies approximately like $1/v$, unlike choppers, if the moderator is kept at a temperature T and if the neutrons of interest have an energy $E_n \gg kT$ or if the moderator temperature T is adjusted in a manner that $E_n \approx kT$. The later case will be adopted for most of the experiments that need monoenergetic neutrons, while the first method will be used for experiments with so-called "white neutron spectra". That the experimental conditions at a pulsed reactor are quite complex can be shown by the following experimental condition: if for an experiment the angular resolution is not the limiting factor, we can gain in neutron intensity at the detector by reducing the moderator thickness and the length of the flight path and still keep the same energy resolution. A gain factor of about 1.85 can be obtained in this condition (2). Unfortunately only a very few experiments can be made where the angular resolution is not a limiting factor. Since we have three variables, the peak neutron flux, source diameter, and burst width to adjust for an optimum experimental arrangement, it is necessary to have some guiding criteria that give some insight in the complex situation. In the following, we will show that there is an interconnection between angular resolution and burst width for time of flight experiment.

II. Neutron beam quality and burst width condition

The aim of neutron scattering experiments is the accurate measurement of one, two, or three parameters: momentum transfer, energy transfer, and intensity. The accuracy of an experiment depends on the angular and energy resolution and on the statistical accuracy of the intensity. These experimental conditions can be described most clearly, according to Maier-Leibnitz (3) in momentum space, Fig. 4. In principle we can now optimize these volume elements for any experiment. However, the experimental conditions will be very often such that one can accept for convenience a reasonable compromise as is done in the classical diffraction spectrometers. There the beam quality of a diffraction spectrometer is determined by the collimation and the mosaic spread of the Bragg crystal. Figure 5 illustrates the situation in four different steps in real and momentum space for a monochromatic source and a perfect crystal, a monochromatic source and a mosaic crystal, a white source and perfect



$$\Delta k^i = \Delta k_x^i = \Delta k_y^i = \Delta k_z^i; \quad V_1 = (\Delta k^i)^3$$

$$\Delta k'^i = \Delta k_x'^i = \Delta k_y'^i = \Delta k_z'^i; \quad V_2 = (\Delta k'^i)^3$$

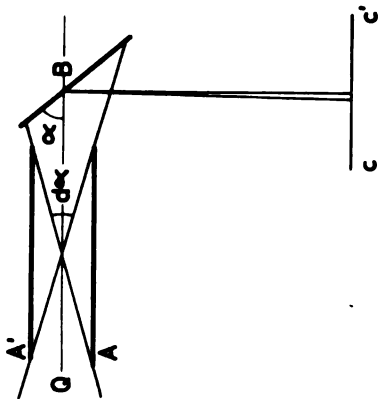
$$V_1 = V_2 = (\Delta k)^3$$

Fig. 4

Momentum space diagram for a coherent inelastic scattering experiment and geometrical presentation of the resolution.

1. Case

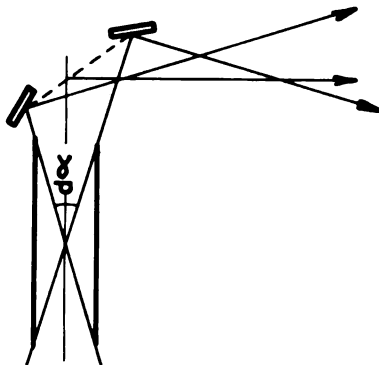
- a) Monochromatic Source.
- b) Perfect crystal, no mosaic spread.



Experimental conditions in real space.

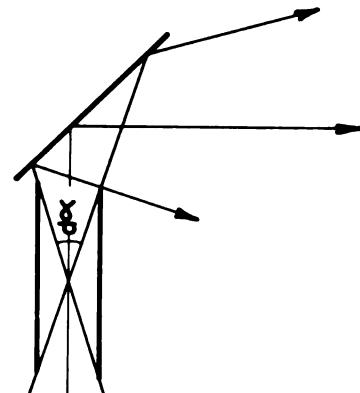
2. Case

- a) Monochromatic Source.
- b) Ideally imperfect crystal, mosaic spread.



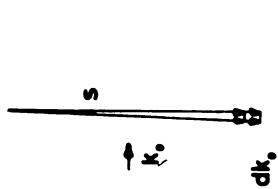
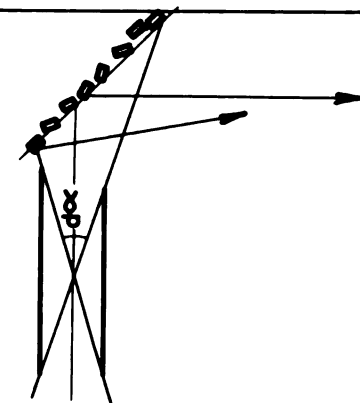
3. Case

- a) White neutron Source.
- b) Perfect crystal.



4. Case

- a) White Source.
- b) Ideally imperfect crystal, mosaic spread.



$S = \frac{\lambda^2 N c E}{h^2 \sin \alpha} \approx 1$ Second of arc
Band width of the monochromatic source dk_0

Neutron distribution in momentum space.

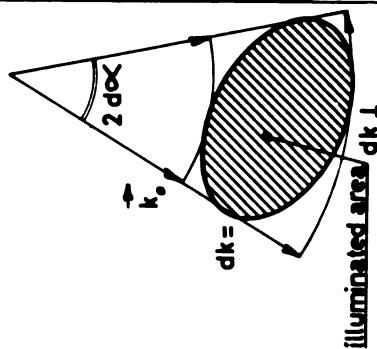
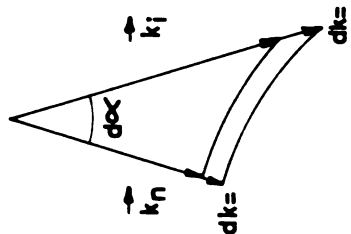
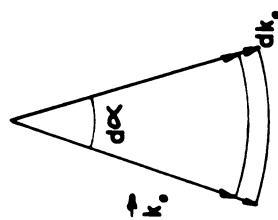


Fig. 5

Momentum and real space diagram

crystal, and finally a white beam with mosaic crystal. The mosaic spread of the crystal is thought to be equal or larger than the collimation angle α . Spreading the k -distribution of case 3 with the corresponding mosaic spread, one sees that the area $dk_{\perp} \times dk_{\parallel}$ will not be illuminated equally well, but that the k -distribution will be rather elliptical in this area.

From Fig. 5 we get immediately:

$$\begin{aligned} (1) \quad dk_{\perp} &= 2k_0 \cdot d\alpha \\ (2) \quad dk_{\parallel} &= k_0 \cot \alpha \cdot d\alpha \\ (3) \quad \frac{dk_{\perp}}{dk_{\parallel}} &= \frac{\cot \alpha}{2} \end{aligned}$$

for $\alpha = 25^\circ$ we obtain $dk_{\parallel} \approx dk_{\perp}$. This means equal error in both directions.

Now, not only diffraction experiments require an accurate measurement of the momentum transfer, but also quite a large number of inelastic scattering experiments, i.e., the determination of the Kohn effect in the dispersion law or the measurement of the diffusive broadening of the quasi-elastic scattered neutrons at small momentum transfer. The high accuracy in K can be achieved best if the volume elements are of equal and of almost cubic size. We assume furthermore, that the collimation is made in a fashion that the K -distribution is equally dense in those volumes V_1 and V_2 , see Fig. 4. Then we obtain immediately for the impinging neutrons the following conditions:

$$\begin{aligned} (4) \quad \alpha &= \frac{\Delta k_{\perp}}{k_0} = \frac{d}{\ell} \\ (5) \quad \frac{dE}{E} &= \frac{2\Delta k_{\parallel}}{k_0} = \frac{2\Delta t_s}{t} \end{aligned}$$

Here α is the angle of collimation, d the neutron source diameter, ℓ the length of the flight path, and V the neutron velocity. From (4) and (5) we obtain:

$$(6) \quad \Delta t_s = \frac{d}{V}$$

For quite a large number of experiments, either elastic or inelastic, the burst time defined in this way will assure the accurate determination of \vec{K} or \vec{q} , the phonon wave vector. Of course this condition reflects only the correct burst time definition for quite averaged cases and not for particular experiments. There are a number of experiments

even in the thermal energy range where one wants to use a somewhat shorter pulse. However, for most elastic and inelastic experiments one will use thermal neutrons, and for them we get an average burst time by using 10 cm diameter beam tube of:

$$\Delta t_s = \frac{0.1 \text{ m}}{2200 \text{ m/sec}} = 45 \text{ } \mu\text{sec}, \text{ for cold neutrons we will need}$$

only a pulse of 90 μsec . The burst time (Ref. 2) can be obtained for thermal neutrons leaking from a cold moderator ($E_n > kT$) or by adjusting the moderator temperature in a manner that the energy of the neutrons of interest is $E_n = kT$. This can also be seen from the following approximate equations:

$$(7) \Delta t_s = T + \tau_R \ln \left(1 + e^{-\frac{T}{\tau_R}} \right)$$

where T = fast neutron injection time

$$\tau_R = \frac{1}{v_n} \cdot \frac{1}{DB^2 + \sum_a}, \text{ the thermal neutron relaxation time}$$

v_n : neutron velocity

D : neutron diffusion constant

B^2 : geometrical buckling

\sum_a : absorption cross section

We shall see that for $T \leq \tau_R$, the burst time $\Delta t_s \approx \frac{1}{v_n}$ and therefore matches automatically with the experimental condition (6). Source pulse condition and experimental requirement are not contradictory but rather are complementary in nature.

Condition (6) can also be used to calculate the beam tube diameter for a given system with fixed Δt_s . For SORA one should have beam tubes of diameter

$$d = 60 \times 10^{-6} \cdot 2200 = 13 \text{ cm.}$$

In analogy to the Bragg reflected neutron beams, the condition (6) can be interpreted as a mosaic spread condition. The burst time should be exactly as long as the neutrons need to fly across the source diameter d .

Pulsed reactors provide the correct pulse width and source diameter, their characteristic data are coherent with the experimental requirement.

Literature

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2. W. Kley, EURATOM-Report 2538 e, May 1965, Proceedings of the IAEA Meeting at Dubna, July 19-22, 1966.
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GLÄSER: Well, after all these arguments I can say that I essentially agree with most of the points of view both of Woods and of Kley. I think the discussion of resolution shows that it is worthwhile not only to ask for higher fluxes, but also to look very carefully for ways of optimizing the equipment. I feel there is certainly a lot to be done to make efficient use of the continuous beam. I do not quite agree with the, let us say, "natural" burst width of 45 μ sec derived by Kley. I think the reasoning is logical for the assumed source; they have a given burst width and now they have to select the proper source area. What is essential is that one needs a fixed solid angle in the experiment and one can arrive at this fixed solid angle by using a smaller source area, smaller distance, and shorter pulses; then one ends up with the same resolution in the case of small samples. The important point in this argument is the size of the sample, which was not mentioned.

Perhaps I can go on to make my point in comparing different types of spectrometers or measuring techniques. I would like to compare TOF versus continuous beam techniques for a steady-state reactor. If the TOF technique can compete with the continuous beam technique, or is even superior, then let us see how a pulsed reactor compares with a steady-state reactor, using the most efficient experimental setup in both cases. Here I would mention that one should take into account the possible improvement in efficient beam use, for example at a continuous beam reactor, by just installing several experiments in line with the same beam tube. In my opinion it is difficult to give figures of merit of general validity; which technique may be the superior one depends upon the type of experiment one wants to perform. Obviously, in any case, the criterion is the effective counting rate at the detector per unit time for otherwise equal conditions, by which I mean equal energy and momentum transfer resolution. The clearest way to estimate intensities is to describe the experiment in momentum space, as was mentioned earlier. The intensity is then essentially determined by the product of the two volume elements describing the uncertainty of the incident wave vector and of the outgoing wave vector. Thus, if these same volume elements are

realized in the pulsed and continuous beam experiments, respectively, the only difference can occur in the time utilization of the beam, the transmission of choppers and filters, the reflectivity of crystals, and the effectively used number of detectors. For example, comparing in this sense the triple axis (TA) spectrometer with a TOF apparatus at a continuous beam reactor, we can introduce a relative figure of merit M as follows:

$$M = \frac{n_{TA} \gamma_E \gamma_{E'} \epsilon_{TA}}{n_{TOF} f_E \tau \nu n_{CH} \epsilon_{TOF}} = \frac{\text{Rate of data collection for TA}}{\text{Rate of data collection for TOF}}$$

n_{TA} = number of third axes

γ_E = reflectivity of monochromating crystals

$\gamma_{E'}$ = reflectivity of analyzer crystals

$\epsilon_{TA}, \epsilon_{TOF}$ = detector efficiency

n_{TOF} = number of detectors in TOF

f_E = transmission of devices in the incident beam

n_{CH} = number of TOF channels

$\tau \nu$ = duty cycle

I apologize for using a figure of merit in spite of my earlier objections against such a criterion. Now if one puts in some typical numbers, let us say for scattering experiments where one is interested in measuring the scattering law for a broad range of momentum and energy transfers, one will end up perhaps with a number of the order of 0.1 taking a reasonable duty cycle, $\tau \nu$, in the denominator. This means that with the TOF technique one can collect the data one needs in 1/10 of the time one would require with the TA spectrometer.

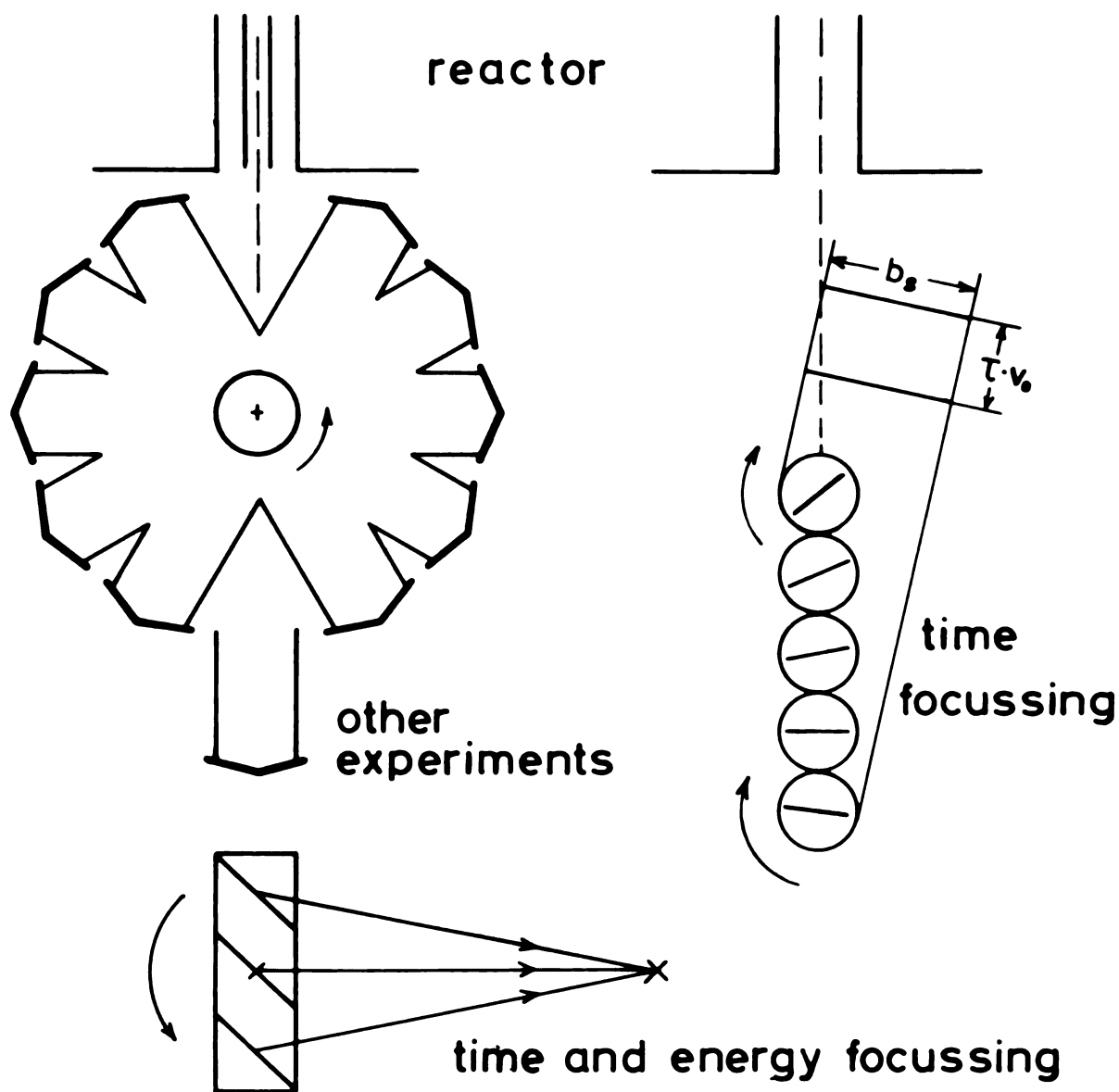
However, if one is interested in measuring just one phonon, as Woods was saying, the figure of merit coming out of this formula would look quite different. It may be something like 2 to 5. This means in this case the TA spectrometer is the superior method. However, one should mention in this case that the argument was with the same small duty cycle and I do not see any reason, if one is interested only in one phonon line or a few phonon lines, why one could not increase the duty cycle considerably in the TOF technique. There is no serious frame overlapping problem. For example, one can think of using just a second detector in the flight path to make a rough energy measurement, and in this case the frame overlapping problem is

ruled out; I think one can end up even in this case with a comparable figure of merit for one-phonon measurements. What one should state in connection with this type of measurement is that there is an increasing interest to get not just one phonon, but to get really a large number of phonons for the whole momentum space. Similar arguments hold if one is only interested in a certain range of energy in the scattering law measurements, for example, the quasi-elastic line in liquids. In this case, the duty cycle can be increased considerably by suppressing overlapping neutrons with filters, mechanical selectors and similar devices.

There may be differences in the momentum space volumes attainable with different methods in energy and momentum transfer for the same final resolution. Although this has to be discussed for each experiment separately, I would just mention one example and that is for phonon dispersion measurements with a TA spectrometer. Focusing methods have been discussed recently which can increase intensity by a factor of about 2 to 10. In this application, in principle, it is the size of the volume element in momentum space which is optimized for a certain resolution, and what I would like to mention is that in the TOF technique one can utilize this type of focusing too. For example, I am thinking of a rotating crystal as a monochromator. A similar example has been given in a contributed paper by Armbruster et al. in which they propose to use back-scattering to increase the momentum space volume elements.

Considering the TOF technique only, I think it is worthwhile to estimate the influence of pulse width and repetition frequency on the intensity optimization. What results is that neither the pulse width nor the repetition frequency is important itself, but that the only important quantity is the duty cycle, that is, the product of these figures. Of course one can increase the pulse width, but one has to increase in the same ratio the time distance between two pulses, and what one still has to do in this case is to maintain constant the solid angle which is involved in the intensity consideration. So here I feel an actual limit is given by the cost of shielding, detector areas and such things, presuming enough space is available. Longer flight paths have, of course, the possibility that one can choose larger scattering samples, but I think it is obvious that the tendency in inelastic scattering work is toward smaller samples. This is not only because sometimes one only has smaller samples, but I think there are also other problems, for example, corrections for multiple scattering and such things which can be handled easier with smaller samples. So in this case I think there remains the disadvantage of higher costs if large pulse widths and flight paths are utilized.

Perhaps I can just make a short comment on the way one can think of making more efficient use of the continuous beam. Some ways of doing this are shown in Fig. 6. We started recently at Karlsruhe to think about using the rotating crystal, which seems to be a proper element to make more efficient use of the continuous beam. I think the first arrangement (upper left of Fig. 6) is known to all of you. There,



SOME FOCUSING PROPERTIES OF ROTATING CRYSTALS

FIG. 6

one simply uses the reflected pulse in every direction from the rotating crystal, which means one makes use of the whole continuous beam in a certain sense; of course one has different energies at different angles. One can also choose several rotating crystals (upper right of Fig. 6) running in phase, for example, to produce a proper time shortening of the pulse and still have a gain intensity which is determined by the dimension of the set of crystals in the reactor beam. Furthermore, in certain cases one can think of using the Doppler effect of rotating crystals to make an energy focusing of the beam; this is shown at the bottom of Fig. 6. So I think there may be a lot to be done in more efficiently using a continuous beam; if this can be done, there may still be some advantage in using a continuous beam compared with a pulsed device with about the same instantaneous flux.

JACROT: It is a difficult position to be the last to speak because then you have to repeat a bit what has been said before, but I will start by saying that I disagree with Kley's statement about the existence of an optimum width of the peak in a pulsed device. I think he is assuming that the momentum resolution and the energy resolution have to be comparable and I think this statement is correct in some cases, but more generally for different programs in physics you may need an extremely good definition in momentum and poor in energy or the other way around. I think the TOF technique is perhaps slightly better from that point of view compared to the crystal technique because there is more flexibility to adjust resolution in one of the parameters while keeping the other constant.

The next thing I want to say is to stress again what was said at the beginning. The important thing is not to have a pulsed reactor or a steady-state reactor but to have a high peak flux and then adapt your experimental technique to the device and not do it the other way. The last point I want to say is the following: I would have said that in the case of inelastic scattering experiments, in which you want to measure the intensity, so far practically no experiment is good enough. I will go a bit farther (maybe making it a bit too strong), but in the field of inelastic scattering up to now we can say in first approximation that it has been proved that it is an extremely good potential technique, but more or less all the experiments which have been done up to now have to be repeated with better accuracy, with a few exceptions. That is to say the question of intensities is extremely important in that field and it has become more and more important now that neutron inelastic scattering has some other techniques which compete with it--I am thinking of the laser techniques.

One of the limitations so far in the experiments with inelastic scattering techniques is a consequence of the fact that the flux of neutrons in the range 0.1 to 1 eV in all the existing devices is extremely small. Because of this the number of experiments in which the incident neutrons used in this range is practically zero. This is a big drawback because in many experiments the neutrons which have to be used must be in this range. If we come back to these experiments, such as the

frequency spectrum of vanadium, I think that it would be much better if they were made with neutron energy loss instead of energy gain, because then you do not have the Boltzman factor and things like this. Another field of application which is very important is the measurement of high levels which you cannot get if you do not use this kind of "hot" neutrons. We can also think of the measurement of phonons in liquids in which you must have the speed of the neutrons higher than the speed of sound in the liquid. I think it is an extremely important thing in any new device to have a way of having high intensities, several orders of magnitude higher than what is available now, of neutrons in the range of 0.1 to 1 eV. That is a question for reactor people and "neutronists" to know what is the best technique to do it. Is it by using a hot source or by using a moderator which does not moderate very well in some part? This is really a question for neutronists and not for us. But I want to stress this point.

PALEVSKY: We will now have questions or comments from the floor.

SPRINGER: I wanted to put a question to Woods. Theoretical arguments show that the resolution of a neutron spectroscopic experiment is approximately proportional to the square root or third root of the intensity of the source (for simple incoherent or focusing experiments, respectively), or with an even higher root. Why do you believe that this argument does not hold practically?

WOODS: I do not doubt that argument at all about the resolution. As a matter of fact the resolution will probably improve much more slowly than the square root of the increase of the intensity; what I am saying is that the amount of physics, which is after all what we are interested in, that you are going to get out of your experiment is not necessarily directly proportional to the resolution which you will be using in that experiment. This depends very much on the particular experiment that you are interested in doing. But I will repeat the point I made earlier that, if the factor of 2 increase in resolution does not improve the overall quality of your experiment by much more than a factor of 2, surely we can do quite reasonable experiments with fluxes of 10^{11} and 10^{12} , which I think you will agree is a ridiculous extrapolation.

SPRINGER: I think that my argument was especially concerned with the measurement of phonons, phonon lifetimes, quasi-elastic scattering--things like this.

WOODS: Well, once again it depends on which particular phonon, or which particular system you are trying to study. It may be in a particular instance that a very modest improvement in resolution will give you the information that you want. You may have two levels which lie very close together--with present resolution techniques you cannot tell them apart--but by just improving your resolution by 10 or 15% you may be able to separate these two levels just ever so slightly, and then you can

learn a tremendous amount about the physics of the system you are trying to study.

BRUGGER: With regard to a comment that Woods made, he pointed out that very few absolute intensity measurements have been made for inelastic scattering. However, some have been made and to an accuracy that shows the physics in the experiment. I would like to point out that most of these have been made with TOF methods since these are more difficult to make than just the energy change, momentum change, and experiments.

BEYSTER: I would like to get back on the matter of pulse widths again. My remarks mainly pertain to Kley's paper. I think that when the energy resolution is the main consideration, then one can today make some fairly clear-cut statements as to what the pulse widths should be for an optimum type of device. Stevens et al. endeavored to do this for the accelerator-booster where the pulse width from the accelerator, the moderator pulse widths and the multiplication of the device are all inter-related in a certain way. Actually, Stevens made fairly pessimistic assumptions on what would happen to the peak flux as one poisoned the moderator. It is not clear that poisoning is the best way to shorten the pulse, but it is what was assumed. For a 1-MW accelerator-booster, one could obtain an increase in flux on the scattering sample by a factor of 3 to 5 more than the mechanical booster. Now this is in the thermal range (below 0.1 eV). If one talks about neutron intensities above 0.1 eV, the pulse width of a mechanically pulsed reactor stays essentially the same, 50 to 60 μ sec. But the pulse width of the accelerator-booster would then be about 5 μ sec total. One thus gains a large factor in figure of merit for a TOF experiment with the accelerator-booster. Probably momentum resolution in the energy range above 0.1 eV is not really vital (I do not know, but I have not heard it mentioned very much in experiments in that energy range), so that there could be a large gain in figure of merit, like a factor of 25 to 50 in that range, by just concentrating the power in narrow pulses rather than spreading it out over 50 μ sec. The other remark I would like to make is essentially the same one I think Gläser made. It seems to me that if the momentum resolution and the energy resolution are related, as Kley mentioned in his talk for certain experiments, then the counting rate is pretty much dependent on duty factor as Gläser pointed out. If you shorten the flight path and decrease the diameter of the moderator, the pulse width which would be required for optimum conditions is narrowed. In addition, the apparatus becomes considerably simpler and less bulky, so that it seems to me that even in this experiment, although there may be no increase in figure of merit, there is an advantage to shorter flight paths.

KLEY: First, I would like to say where we really do not disagree and then I want to make a few remarks about simple apparatus. I have derived this formula, $\Delta t = d/v$, in order to have some idea about burst time requirements. It is clear that it can

only represent a compromise for the large number of special experiments and cannot be considered as a general formula that covers every possible experimental condition. However, as everybody can see, the v stands in the denominator. Therefore, if one wants to use neutrons of an energy greater than 0.1 eV, one needs much shorter pulses than can be produced by pulsed reactors. A booster system is much more favorable. So this formula does not disagree with your statement whatsoever.

Concerning the other statement, also made by Gläser, that experiments with thermal neutrons having shorter pulses and smaller source diameters and correspondingly a shorter flight path are simpler and cheaper, I can only present the following arguments and conclusion. It is true that the flight path of a TOF experiment can be reduced corresponding to reduction in the burst width and still obtain the same energy resolution. But now I would like to see what is simpler: an experiment using a pulsed reactor or a pulsed reactor with injector that delivers shorter thermal-neutron pulses. In the first case, one needs a relatively large source and a long flight path, corresponding to the long burst time of pulsed reactors, that will really add only to the expenses, but not to the complexity of the facility operated with a pulsed neutron source. In the second case, one has to buy a linear electron accelerator to make the pulse shorter, then one can really use a smaller source diameter to get the same angular resolution and, of course, the same intensity that is defined by the angle α . So I totally disagree with the statement that the second type of apparatus is cheaper and simpler.

In a previous discussion, several people have claimed that the formula $\Delta t = d/v$ is not giving the right burst time definition. I just want to point out that not only I, but all the people here doing neutron diffraction spectrometry, should object because they really define neutron beams as I have defined them in analogy for TOF experiments. Using shorter neutron pulses than adapted to the angular collimation means, in analogy, to work with wide collimation and very small mosaic spread. As far as I know, this is not done for most of the diffraction work.

MAIER-LEIBNITZ: I see really no reason for serious disagreement in all we have heard. It may be that each man has emphasized a little too much one point and has forgotten another one, but I am sure that if this discussion went on a little longer and everybody had time to think we would all agree on every single point we have discussed. I think that we might summarize what we need to do for good experiments. First I think you should have something like the famous ω (energy loss) and κ (momentum transfer) diagram. You must look at that diagram for the various things you want to do. If you want to do quasi-elastic scattering in liquids, there is no reason to have good κ resolution; you want good ω resolution because you know the widths of the quasi-elastic lines vary slowly. If you do the phonon dispersion curves and you look at that part of the curve about which you know something, then you can use a focusing method in order to get optimum intensities. You can really take that diagram along

with your physics programs and put in the resolutions you want for κ and ω , and then you start looking for the best experiment. This is really the process through which you go when you do experiments. I think you can now go through that process using a wider variety of instruments and experimental methods.

Now we come to the question that interests us professionally, do you want a pulsed reactor or a continuous reactor? The pulsed reactor has certain properties-- it has a maximum flux, which in the cases we have seen is about the same or in the case of SORA perhaps three or four times higher than the flux of the best continuous high flux reactor. However, you have certain properties which are hard to change; you have a pulse duration that is hard to change; and you have a repetition time that is quite often even harder to change. Then you have a certain maximum solid angle you can get from your source and you have the problem of background which will be more favorable than in a continuous reactor. Now on the other hand, you can use the pulsed reactor as an energy selection device instead of a crystal. This energy selection may be better or not so good as the energy selection you can get with continuous reactors. But if you take all the points into account, there is no doubt that a valid comparison between pulsed and continuous reactors will come out. It seems that, roughly speaking, except for the possible solid-angle argument in cases that are favorable for pulsed reactors, they may be exactly as good as continuous reactors and then they may have advantages due to background or maybe due to larger solid angles, etc. But there are cases when it is just the pulse repetition time and things are not as favorable as for continuous reactors, so I do not think there are any serious problems.

The last thing I want to mention is that we have tried to make up lists of applications of things you want to do, and one thing that has given us most difficulties in devising good experiments is those experiments in which you want relatively high neutron energy. This refers to cases when you want finite energy loss or gains at small momentum transfers. In molecular crystals and optical vibrations this should be really quite important, as well as in other cases where you want high absolute energies, for instance, in magnon scattering and things like that. For whatever you have in high flux continuous or pulsed reactors, I think it would be worthwhile thinking about that problem more carefully than has been done before. It seems to be a very difficult problem.

SCHULT: This question is directed to Jacrot. Do you think the remark that almost all experiments have to be repeated with better accuracy may also be made at another SINS meeting, say after five years?

JACROT: I do not think that this is strictly correct because there exists already a certain number of experiments in which the level of accuracy, which one must obtain to have useful information, has been obtained, for instance, like seeing the equation of the roton in liquid helium and maybe in some cases of phonons. In the case of

phonons, I think, if the accuracy is increased by a factor of 2 or 3, the experimental data would be good enough so that the theory can rely completely on it, but this state has not been completely achieved.

WOODS: I think, if you can get a factor of somewhere between 2 and 10 in resolution or in accuracy of an experiment and then if you repeat it under those conditions, most of the time you are going to discover something that is new and worthwhile. I think that for most experiments if you improve the accuracy of the measurement by a factor of 10 you will discover something--maybe some side effect--which has made the repeating of that measurement worthwhile.

JACROT: But my point was raised to stress the big difference with the case of neutron diffraction in which, in many cases, the problem is completely settled and you do not have to come back to it. We do not need improvement in the resolution except if we are really accomplishing something new, some new physics.

SCHULT: I agree here completely with what Jacrot said. I discriminate between a repetition of a measurement and an improvement. It is definite that we always will have new problems and we will wish to improve the situation, but I do not consider this as a repetition. I would say you repeat a measurement if you find that the first result does not tell you anything, so what Jacrot said is, "Is the physics you learn from the experiment enough?" My question was, if we get a certain kind of improvement in flux and so on, is this enough to really get physical information from the data or should we maybe perform only those experiments where we can expect this amount of data.

PALEVSKY: Well, I should like to take a few minutes now as Chairman to impose my views. We have had a little bit of experience at Brookhaven because the high flux reactor, HFBR, is now running, and in making the transition between our graphite reactor and the high flux reactor the thing that has really been hard for us to anticipate is the extreme large size of shields that are necessary. As many people have said, when you use TOF techniques at a steady-state reactor you are throwing away 99% of that beam and if you have high fluxes coming out that means you have to contain that beam, otherwise you are going to ruin all the other experiments as well as your own. So the shielding becomes a very big problem. Therefore, from my fortunate position at Brookhaven, the next step to me would seem to be a flux something like 10^{16} or 10^{17} . Then it seems to me that the only reasonable kind of device that will give these fluxes will be pulsed devices because, in steady-state devices, you will either burn up the cores too quickly or you will be just overwhelmed with trying to shield the beam.

HASTINGS: With respect to the background, I think that for the triple axis spectrometer, which is also now operating at the HFBR, that our background is much lower than we had expected. As a matter of fact, it is down to less than one count a minute with the reactor flux at about 10^{15} , and I do not see that we can ask for

very much better. So I anticipate that going up a factor of 10 or 100 in a steady-state reactor we could still conveniently handle the background problem.

WILKINSON: I would like to make a few brief comments about a new TOF technique for inelastic scattering which is being developed at Oak Ridge. (I would just like to note that perhaps this indicates we do not always stick with conventional techniques, unless they happen to suit our purposes.) This technique is being developed almost exclusively by Herbert Mook; we think it is very simple and has a tremendous amount of flexibility and actually can incorporate many of the features which have been suggested as requirements by our Panel members. The technique is briefly this: one takes a magnetic reflection from a magnetic crystal (a reflection which is essentially 100% magnetic) and then, by pulsing that crystal with a magnetic field along the scattering vector, you can cut off the reflection, so that you actually do your pulsing using a crystal; you can flip these spin systems with extreme rapidity in the order of 10^{-10} sec. This means that you can get good time resolution in your pulses and about 2 μ sec have been attained. If you want wider pulses, you merely dial it in; if you want to change repetition rate, you also dial it in. It has a tremendous amount of flexibility which I think can solve many of the problems which our Panel has suggested.

HASTINGS: We are doing something similar for polarized beams. We are using the rf flipper, which is normally used to change the orientation from up to down, and frequency modulating it so that we can also get a modulated beam of polarized neutrons with a variable pulse width and a variable duty cycle.

SESSION VI PART C - PANEL DISCUSSION

ON

NEUTRON AND FISSION PHYSICS

Panel Members: A. Michaudon (Chairman)
P. Armbruster
L. Bollinger
J. Milton
V. Sailor

Secretary: R. Leachman

MICHAUDON: In this Panel, we shall deal with five topics:

1. Fission induced by thermal neutrons.
2. Fission at all energies except thermal.
3. Total cross sections measured with small samples, and general considerations of slow neutron spectrometers.
4. Measurements using polarized neutron beams and polarized targets, and an experiment to study the neutron-neutron interaction.
5. Fast neutron spectrometers.

Each topic will be discussed by one Panel member and then among the Panel members and by those in the audience who want to ask questions or make comments. The first topic--fission induced by thermal neutrons--will be treated by Armbruster.

ARMBRUSTER: The main point of the discussion this morning, pulsed or steady state reactors, is covered for the thermal-fission work by one statement. Steady-state, high-flux reactors are needed for the thermal-fission work. Pulsed reactors may well be of some use in resonance work and that will be covered by Milton in his comments.

It is necessary to introduce an energy of ~ 5 MeV into a nucleus, if it is to fission, to overcome the fission barrier (shown schematically in Fig. 1). The nucleus then moves over the barrier, and what one studies mostly in thermal-fission work are the late stages of fission--the scission-point configuration (that means the breakup process of the nucleus into two nuclei) and the properties of the primary fission products (beta-decay characteristics and related features). Thermal fission is not very well-suited to a study of the properties of the saddle-point configuration, or as it is often called, the transition-state configuration, because the energy you introduce by thermal fission is about one MeV higher than the saddle-point barrier. Thus, in thermal fission we are in general well over the lowest fission channel.

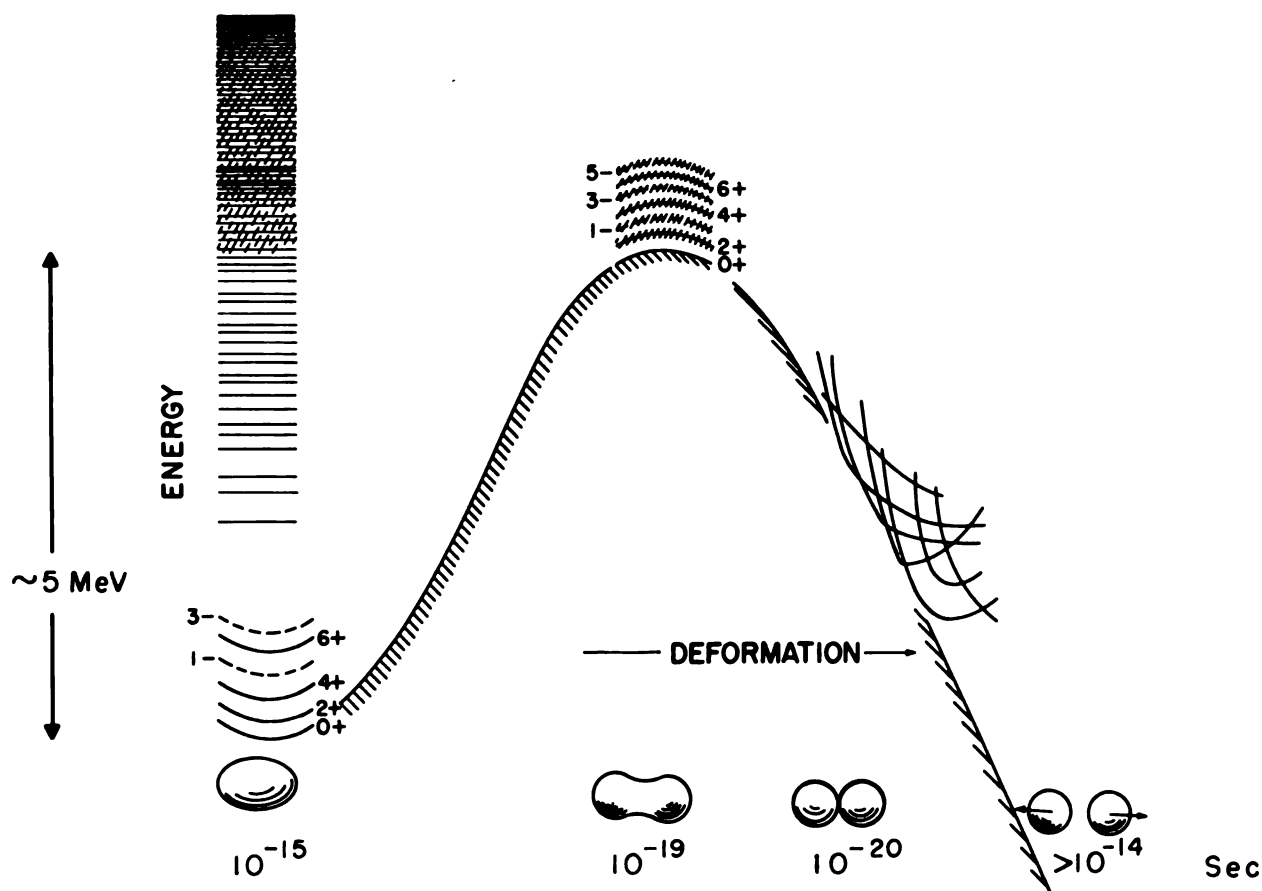


Fig. 1. Barrier model for fission of even-even nuclei. Times shown characterize the typical times at each stage: pre-saddle, saddle-point configuration, saddle-to-scission, and separated fragments.

I have written here some properties we want to measure as functions of the primary (pre-neutron) masses and energies of the fragments, m^* and E_K^* .

Yield Distribution, $W(m^*, E_K^*)$

Distribution in Average Primary Charge, $\bar{Z}_p(m^*, E_K^*)$

Fission-Neutron Distributions:

Number, $N_n(m^*, E_K^*)$, and Average Energy, $\bar{E}_n(m^*, E_K^*)$

Fission Gamma-Ray Distribution:

Number, $N_\gamma(m^*, E_K^*)$, and Average Energy, $\bar{E}_\gamma(m^*, E_K^*)$

The most important property seems to be the probability distribution of the masses and kinetic energies of the fission products for the thermal-fissioning isotopes. This quantity is fairly well-known today, and measurements of these quantities are now available including the low-yield fission products in the mass valley. But we will have to study all the other quantities, that is, the primary charge of the fission products, the number of neutrons, the energy of the neutrons, the number of gamma quanta, and the average energy of the gamma quanta. To study in detail the dependence of these quantities on fragment mass and kinetic energy, it is necessary to increase the flux as much as possible. All these measurements are coincidence measurements between the two fission fragments.

To do these coincidence measurements, we have to use an external beam, and the relations we use for the mass determination are as follows:

$$m_1^* v_1^* = m_2^* v_2^* \quad (1a)$$

$$m_1^* + m_2^* = M, \quad (1b)$$

that is the momentum conservation and the mass conservation between the two particles. The masses and kinetic energy of the fragment are determined from a measurement of the velocities and the energies. For all these measurements, thin targets are needed ($50 \mu\text{g}/\text{cm}^2$). The total number of fissions we detect is proportional to the integral of the total flux over all energies times the cross section, which goes as v^{-1} .

$$A = C \int \Phi(v) \frac{1}{v} dv. \quad (2)$$

I will return to this rather important point shortly.

The other possibility for determining the masses of the fission products is to use mass spectrometers. From a mass spectrometer you can determine the velocity and the ratio $\frac{e}{m}$ of ionic charge to mass by a combination of electric and magnetic fields.

$$v = \frac{F}{B} \frac{\rho_e}{\rho_m}, \quad \frac{e}{m} = \frac{F}{B^2} \frac{\rho_e}{\rho_m^2} \quad (3)$$

The mass resolution of such a device has to be of the order of 800 and the energy acceptance as large as possible. Here again, Eq. (2) holds. The total number of fissions is proportional to the integral over the flux times v^{-1} , as all these targets are still thin ($\sim 400 \mu\text{g}/\text{cm}^2$) compared to a black target.

Our problem is now to optimize our beam hole for the external-target experiment in such a way that we get the highest number of fissions. Here, I tried to give a comparison between neutron guide tubes and normal beam tubes with collimators, both for the cold source and for a thermal reflector. To do the comparison I have to make some requirements to the beam quality. I required the ratio between epithermal neutrons and thermal neutrons to be 10^{-5} and the gamma flux, not to be larger than 100 mr/h. I compared the number of fissions in a 2-cm diameter fission source for a reactor with a flux $\phi = 10^{15} \text{cm}^{-2} \text{sec}^{-1}$ at the inner end of the beam hole. This will give, for an optimized beam tube with collimators, 2×10^6 fissions/sec. The comparison gives

Beam Tube Optimized		Neutron Guide Tube	
Thermal	Cold Source	Thermal	Cold Source
1	12	0.4	5

I only give the ratios to the thermal beam hole which we normally use. Such a beam hole looking at a cold source gives a gain of a factor of 12. The gain using a neutron guide tube at a cold source compared to a thermal beam hole is a factor of 5. This increase results from two effects. One is that the transmission of a neutron guide tube is increased, if you go to cold neutrons, and the other is a gain for both the neutron guide and the normal beam tube coming from the v^{-1} dependence of the cross section.

With reactors of the next generation, or with the Brookhaven HFBR, and optimized equipment, a fission source strength of the order of 1 mCi may be obtained. This is at the limit for the coincidence experiments which are restricted by the time resolution of the coincidence circuits. If the strength of the fission source gets too high, the number of accidental coincidences relative to true coincidences becomes too large. With a time resolution of 10^{-9} sec and a ratio of accidental to true coincidences of 0.1, a source strength value of the order of 3 mCi is obtained. This limit is available now with fluxes of $10^{15} \text{n}/\text{cm}^2 \text{sec}$. Fluxes higher than $10^{15} \text{n}/\text{cm}^2 \text{sec}$ do not seem useful for coincidence experiments in external beams.

The next topic I want to discuss is the possibility of having a device at a high-flux reactor which produces the short-lived fission products in sufficient amounts to study the properties of the short-lived isotopes with half-lives down to a few tenths of a second. The number of nuclei which are produced in fission and which have very short half-lives is very high, of the order of 100. A few of them have known half-lives, but for many even the half-lives are unknown. Nuclear spectroscopy of the short-lived isotopes will be the main application of such an isotope production device.

A mass separator at the high-flux reactor would, furthermore, give us the most complete information on the charge distribution in fission. The number of beta particles as functions of the mass of the fragments and the kinetic energy of the fragments can be counted. Our choice for a spectrometer was a focusing parabola spectrometer which compares to the old Aston parabola spectrograph as a modern camera to a pinhole camera. Most of the work we have done until now is to study the focusing properties of such a parabola spectrograph. In Table I different devices for on-line separation of fission products are compared. One is a Cohen-type gas-filled spectrometer which was built for the first time 8 years ago at Oak Ridge. Others are a Herzog-Mattauch spectrometer and a focusing parabola spectrometer. The point which is most important are the numbers in the last column. For a parabola spectrometer with a resolution of 800, we will obtain about 1 μCi with fluxes of 10^{15} n/cm²sec, for a 6% yield product. With a gas-filled separator you can obtain sources which are nearly three orders of magnitude higher. Its poor resolution may partly be compensated by the use of high resolution γ detectors; the low-resolution device coupled with sensitive high-resolution γ detectors will still be useful in the future.

The ion optics of a parabola spectrograph are shown in Fig. 4.3 of Paper VI.E. The parabola spectrograph we plan is a combination of electric and magnetic sector fields, and the two deflecting fields deflect in perpendicular directions. You get focusing properties in the radial direction by the magnetic field and focusing properties in the axial direction by the electric field. Calculations of the ion optics (including higher order aberrations and fringing fields) have been done with a group at the University of Giessen. More information is given in our contributed paper. A first design is shown in Fig. 4.4 of Paper VI.E. The spectrometer is very large: 12-m horizontal dimension, and a height of 7 m. In size, the total installation is hardly comparable to any available spectrometer.

To conclude, I want to make four statements: (1) A more complete knowledge of the last stages of fission will be obtained with better data from the correlation work I mentioned. (2) The mass separator for the fast fission products will give us for the first time short-lived, neutron-rich isotopes in amounts high enough to study decay schemes reasonably well. (3) Use of cold sources for the beam-hole experiments in fission is recommended, because you gain the factor of v^{-1} . (4) Higher fluxes in excess of 10^{15} n/cm²sec, I think, are not very useful for these experiments because in the coincidence work the usable source strength is restricted by time resolution. For the mass spectrometer, the cooling problem of the fissile source is getting too severe; and the burn-up of the fissile source is so rapid that we get burn-up times of the order of three days, this requires a source change every few days.

MICHAUDON: I shall give some general features of slow neutron spectrometers as well as some considerations on neutron total cross-section measurements with small samples.

TABLE I
Comparison of different fission product separators
($\phi = 10^{14}$ n/cm²sec, $\sigma_f = 580$ b)

type of separator	Thickness of fissionable layer in reactor [mg/cm ²]	luminosity [cm ²]	mass resolution $\frac{M}{\Delta M}$	$\frac{N(e)}{\Sigma N(e)}$	$\frac{N(E)}{\Sigma N(E)}$	effective source strength	Purity of source light heavy products [%]	source strength for 6% yield isotope [sec ⁻¹]
gas filled separator, () data for FRJ-2 separator	1,6 (0,3)	$20 \cdot 10^{-5}$ ($1,7 \cdot 10^{-5}$)	25 (16)	1,0	1,0	0,6 (0,08)	25 (15)	$900 \cdot 10^3$ ($3,7 \cdot 10^3$)
Herzog Mattauch spectrometer, () data for FRM separator	0,4 (0,3)	$10 \cdot 10^{-5}$ $0,5 \cdot 10^{-5}$ ($0,35 \cdot 10^{-5}$)	150 800 (150)	0,15	0,09	1,0	50 95 (50) (95)	$2,4 \cdot 10^3$ $0,17 \cdot 10^3$ ($0,08 \cdot 10^3$)
focus. parabola spectrometer	0,4	$10 \cdot 10^{-5}$ $0,5 \cdot 10^{-5}$	150 800	0,15	0,76	1,0	50 95 99	$20 \cdot 10^3$ $1,5 \cdot 10^3$

I should like to compare two machines for total cross section measured with a small sample: a chopper installed at a high-flux reactor (e.g., HFBR at Brookhaven) and an electron linear accelerator similar to the one which is installed at Saclay. Suppose that the resolution is determined by the burst width of the chopper in the first case and by the moderation time in μsec and where E is in eV

$$\Delta t = \frac{1.8}{\sqrt{E}} \quad (4)$$

in the case of the linac. Then calculation shows that the ratio of the average counting rate $\langle N_{\text{acc}} \rangle$ in the case of the linac to the counting rate $\langle N_{\text{ch}} \rangle$ of the chopper is:

$$\frac{\langle N_{\text{acc}} \rangle}{\langle N_{\text{ch}} \rangle} = 0.1 \sqrt{E} \frac{P_{\text{acc}}^p}{P_{\text{HFR}}} \frac{S}{S_s} \Delta t_{\text{ch}} \frac{f_{\text{acc}}}{f_{\text{ch}}} \quad (5)$$

where

- P_{acc}^p = Peak power of the linac in MW
- P_{HFR} = Average power of the high-flux reactor in MW
- S_s = Area of the slits of the chopper (a few mm^2)
- S = Area of the sample, $S \geq S_s$
- Δt_{ch} = Burst width of the chopper in μsec
- $f_{\text{acc}}, f_{\text{ch}}$ = Repetition frequencies.

A high-flux reactor with a good chopper is now compared with a good linac. In the case of the chopper with the high-flux reactor we use

$$\begin{aligned} P &= 100 \text{ MW} \\ \Delta t_{\text{ch}} &= 1/4 \mu\text{sec} \\ f_{\text{ch}} &= 2 \times 10^3 \text{ cps.} \end{aligned}$$

We suppose that the area of the sample is equal to the area of the slits, $S = S_s$, which is favorable to the chopper. In the case of a good linac, taking

$$\begin{aligned} P_{\text{acc}} &= 200 \text{ MW (100 MeV, 2A)} \\ f_{\text{acc}} &= 2 \times 10^3 \text{ cps} \end{aligned}$$

then the ratio of the counting rates is

$$\frac{\langle N_{\text{acc}} \rangle}{\langle N_{\text{ch}} \rangle} = 0.05 \sqrt{E} \quad (6)$$

This means that the two machines are comparable at an energy of 400 electron volts the chopper being better below and the linear accelerator being better above.

If we now compare a chopper installed on a continuous reactor (CR) to a chopper installed on a pulsed reactor (B), what would be the answer? We can say that, for small samples, the ratio of the counting rates would be

COMPARISON OF NEUTRON VELOCITY SPECTROMETERS IN OPERATION

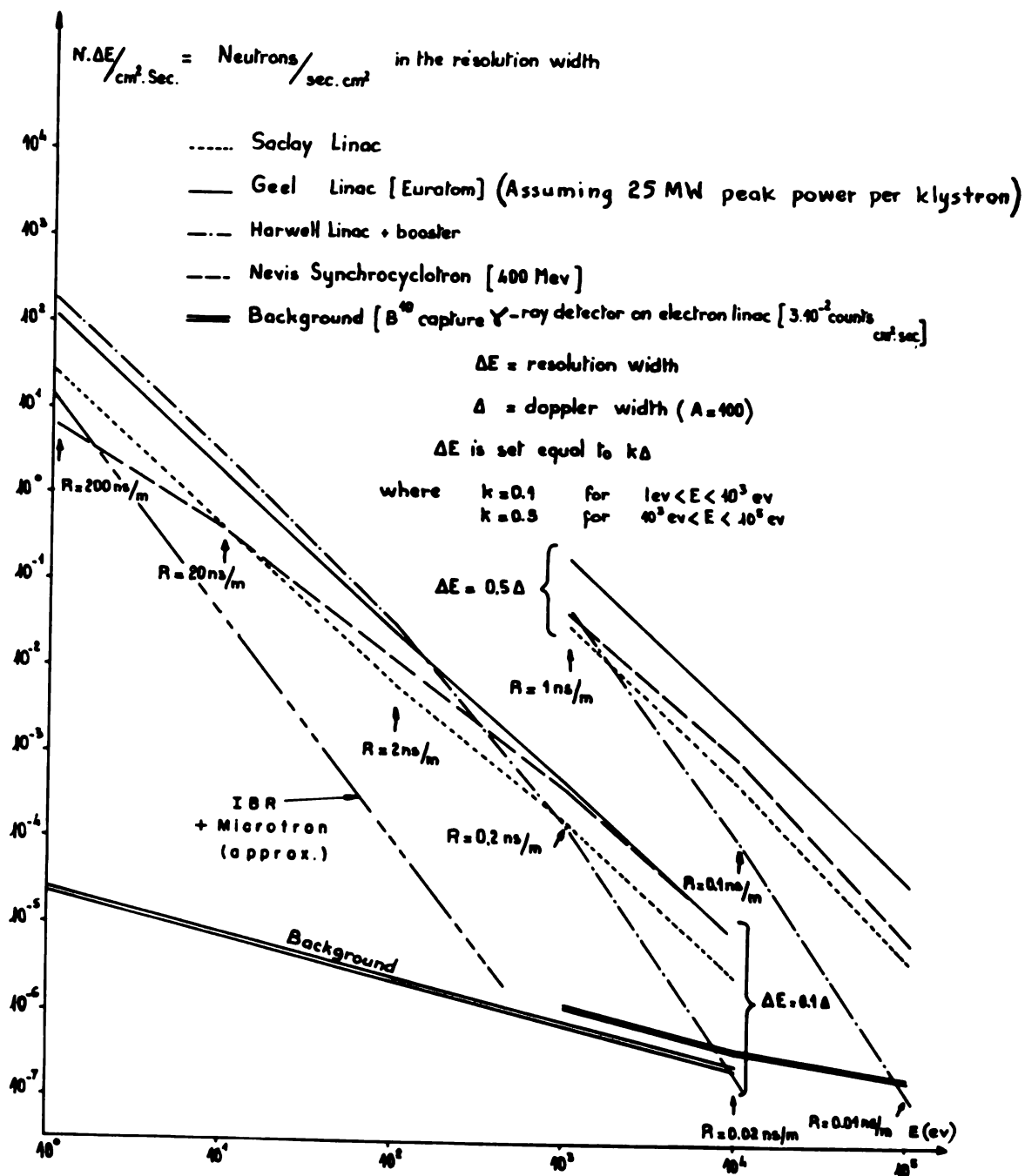


Fig. 2

$$\frac{\langle N_{CR} \rangle}{\langle N_B \rangle} \geq \frac{P_{CR}}{P_B} \times \frac{f_{CR}}{f_B} . \quad (7)$$

Perhaps the geometry is more favorable in the case of the continuous reactor than in the case of the pulsed reactor, and so this gives the inequality. We have for SORA a peak power of 640 MW with a repetition frequency of 150 cps and for the continuous reactor 100 MW and a repetition frequency which is the 2000 cps of the chopper. Therefore, this shows that this continuous reactor is a little better than the pulsed reactor;

$$\frac{\langle N_{CR} \rangle}{\langle N_B \rangle} \geq 2 . \quad (8)$$

On the other hand, because the repetition frequency is lower in the case of the pulsed reactor the signal over the background is better with the SORA plus injector.

Some slow-neutron velocity spectrometers are compared in Figure 2. Slow neutrons are defined here as having an energy between 1 eV and 100 keV. The number of counts per square centimeter per second in the resolution width at the detector position is plotted versus the energy. The resolution width is equal to 0.1 times the Doppler width between 1 eV to 10 keV (lower left lines) and 0.5 times the Doppler width between 1 keV and 100 keV (upper right lines). The background is included to establish how much the signal exceeds the background. First we can see that for low energy the Saclay linac and the Nevis synchrocyclotron are quite comparable. The slopes are a little different because in the case of the Nevis synchrocyclotron the number of neutrons per burst is constant, but in the case of the linac the number of neutrons can be adjusted to the moderation time. Therefore, the counting rate rapidly becomes higher at low energy, but is lower at higher energy. Where the two machines are comparable in neutron yield, the power in the Nevis proton synchrocyclotron is 200 W, but in the case of the Saclay linac it is close to 1 kW.

Table II gives some information concerning the calculations. We suppose that in the case of the linac 10% of the fast neutrons were used. In the case of the

Table II. Bases of Comparison Between Slow Neutron Spectrometers.

Accelerator	Percentage of Fast Neutrons Used	Comments
Linac	10	
Phermex (470 kW)	10	Probably too optimistic
Linac, plus booster (x 10)	3	Geometry
Synchrocyclotron	2.5	Geometry plus spectrum and angular distribution of fast neutrons.

S.O.C. (micropulse)	1	Geometry plus spectrum and angular distribution of fast neutrons.
S.O.C. (storage ring)	0.1	Geometry plus spectrum and angular distribution of fast neutrons plus efficiency of the storage ring.

synchrocyclotron we assume that it was 2.5% due to the geometry, the spectrum, and the angular distribution to the fast neutrons. (The spectrum is harder in the case of the synchrocyclotron, and the neutrons are peaked forward.) In the case of the linac plus booster, we take 3% because of geometry. (The booster provides a factor of 10 in total source strength, but it loses a factor of 3 because of geometrical considerations.)

Above 100 eV the Harwell booster (Fig. 2) drops more sharply than the others because at this energy the moderation time is roughly equal to the relaxation time of the booster. Therefore, there is quite a loss in the resolution. The solid line is the Geel linac (assuming a peak power of 25 MW per klystron), which is a little better than the Saclay linac. I should mention some other figures, the IBR with microtron at Dubna at 1 eV is 14. This means it is roughly equal to the Saclay linac at 1 eV but the line drops very sharply. Therefore, it does not compete at all for high resolution experiments at higher energies.

Figure 3 shows a set of proposed neutron velocity spectrometers including: SOC storage ring pulses, SOC micropulse operation, a good electron linac (say 100 kW for long pulses), improved Columbia synchrocyclotron, a Phermex of 470 kW, and background as shown. In order to carry out the calculations I assumed that for Phermex the production of moderated neutrons was equal to that of the linac (but I think that is too optimistic because in order to dissipate 470 kW one needs a big target and therefore one certainly will lose just by geometrical considerations). In the case of the SOC, conditions in Table II were taken. For the SOC micropulses, 1% was taken for the same reasons as with the synchrocyclotron, but because the spectrum is still harder, and the beam power higher, 1% was taken instead of 2.5%. And in the case of the SOC plus a storage ring, 0.1% was taken. That means 100 less than a linac because, in addition to the effects mentioned above, a safety factor of 10 was taken for the efficiency of the storage ring. In Fig. 3, SORA plus an injector gives 7.5×10^3 at 1 eV, and the line for SORA plus injector drops off rapidly with energy. We can see from this set of curves that only at lower energies is the linac below the SOC micropulse, so the SOC micropulse is not better than a good linac. This is not true of the improved Columbia synchrocyclotron, which is roughly a factor of 10 above that of a good linac. Phermex competes very well with

COMPARISON OF PROPOSED NEUTRON VELOCITY SPECTROMETERS

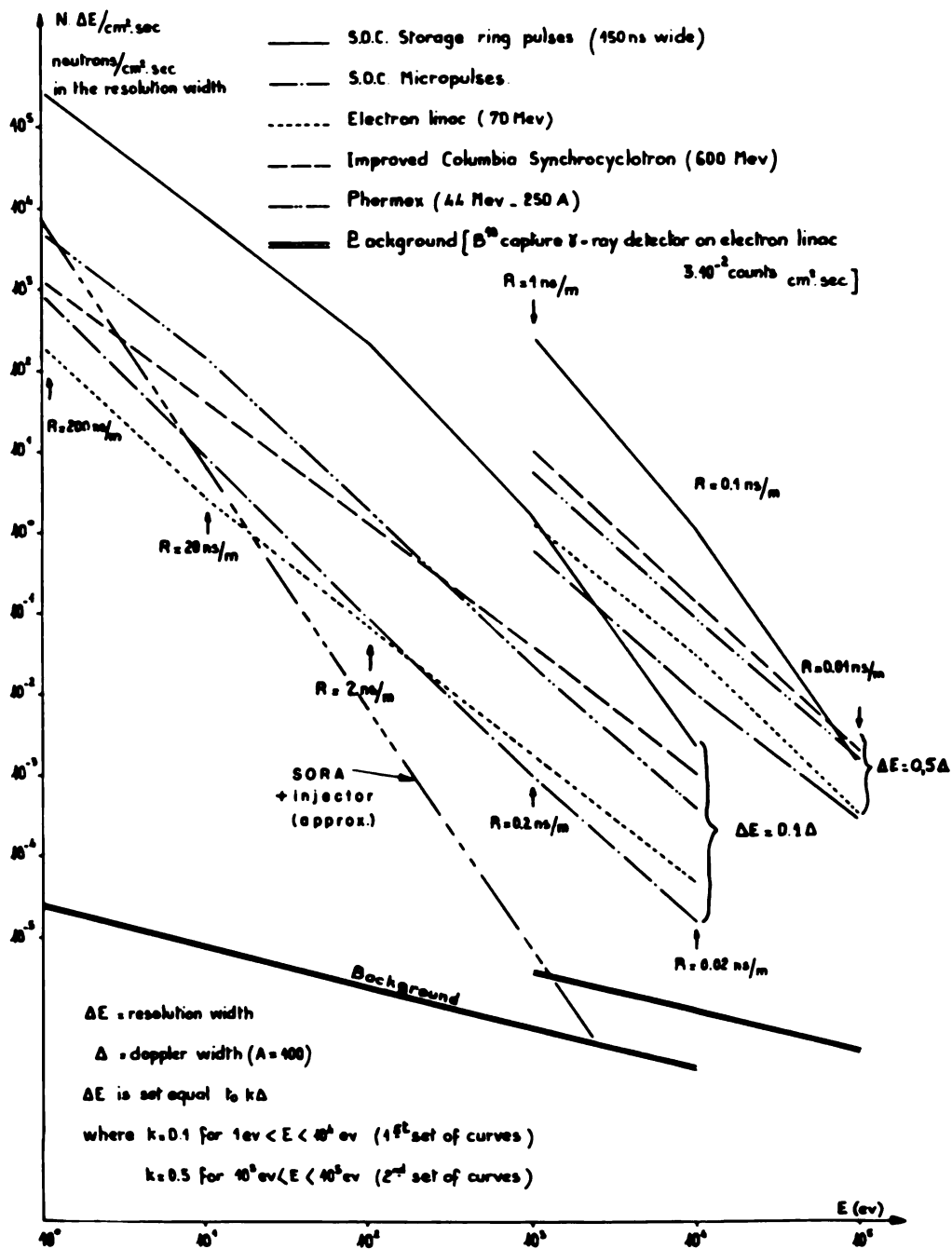


Fig. 3

COMPARISON OF NEUTRON VELOCITY SPECTROMETERS IN OPERATION

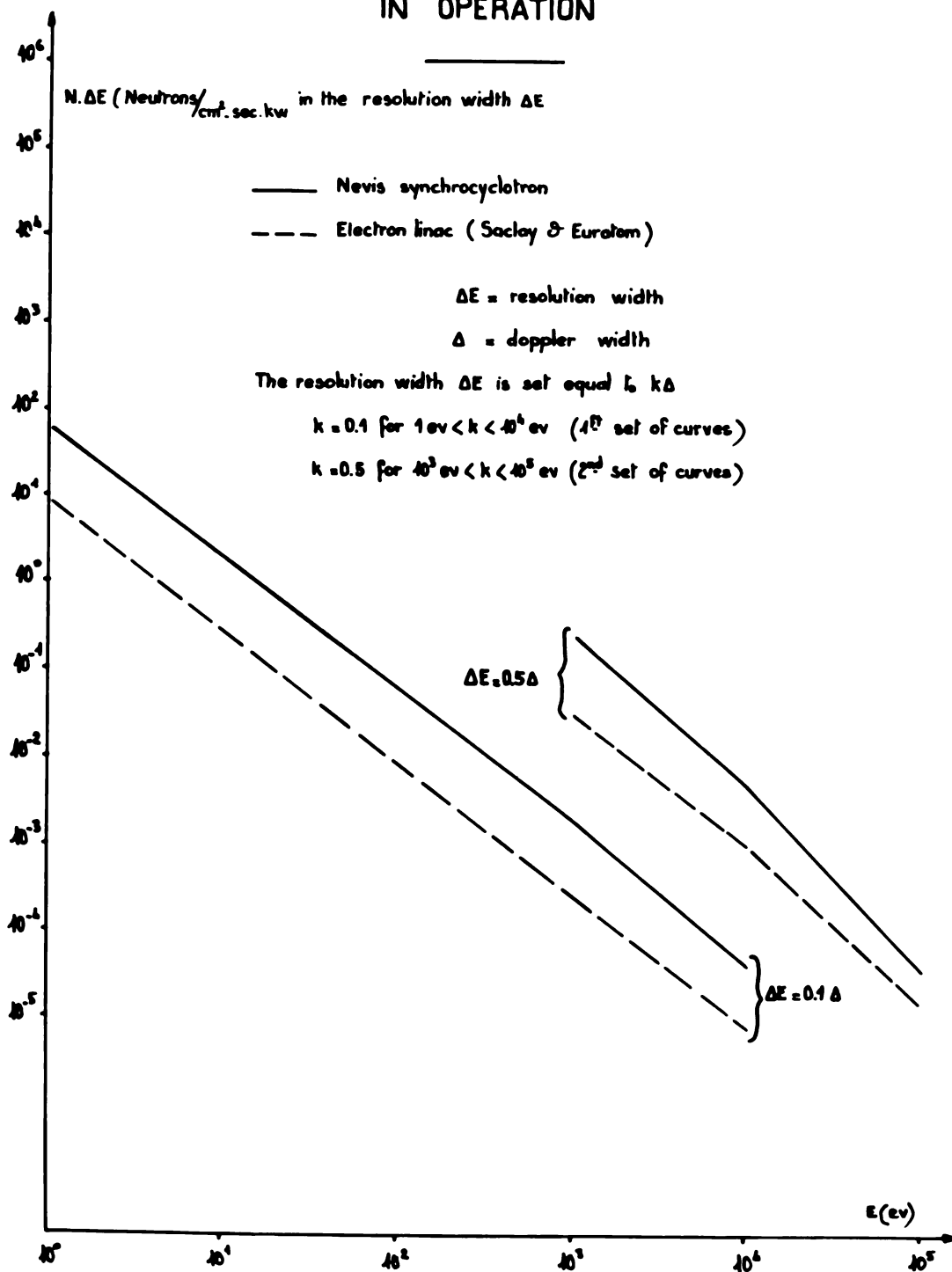


Fig. 4

COMPARISON OF PROPOSED NEUTRON VELOCITY SPECTROMETERS

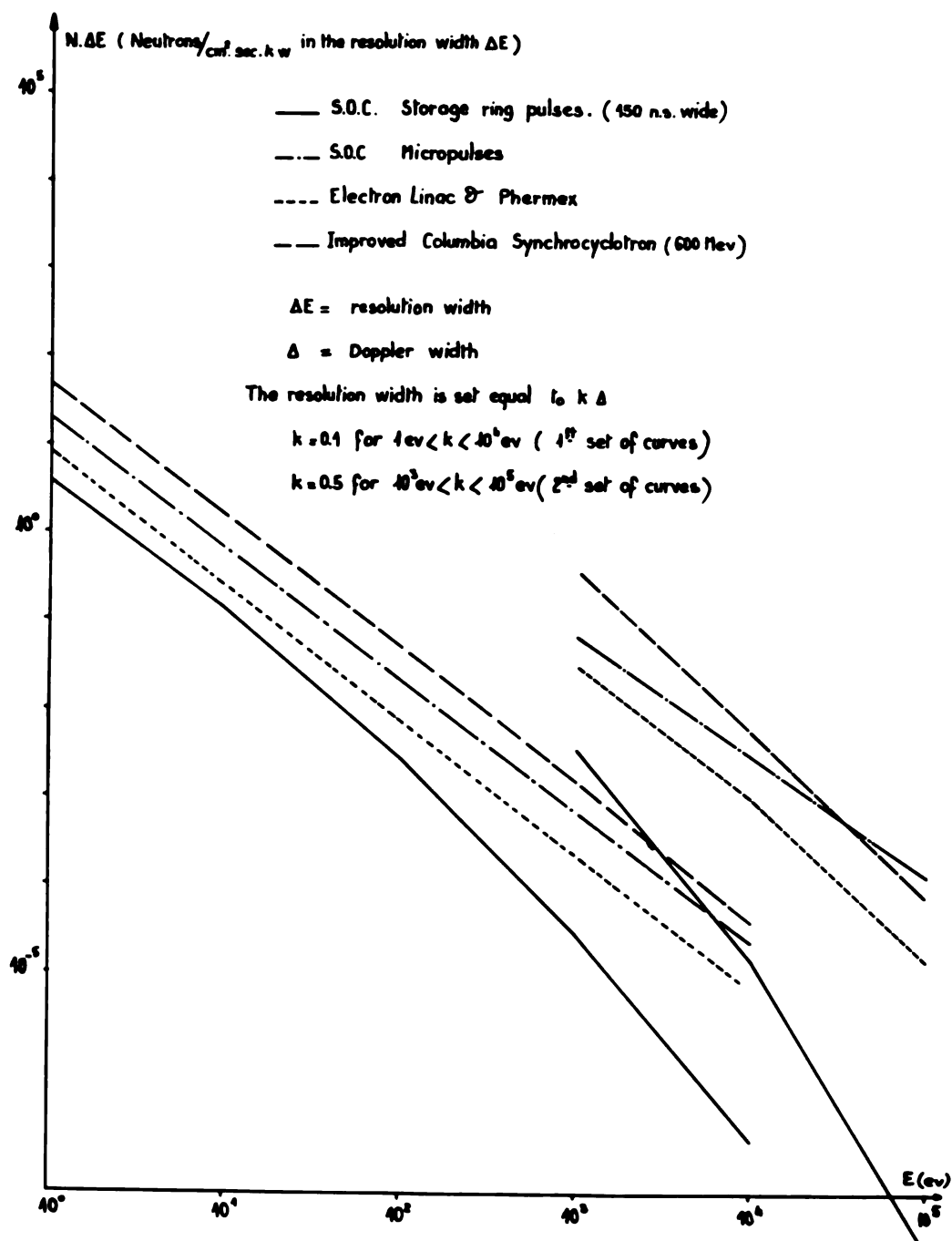


Fig. 5

the Columbia synchrocyclotron, but, as I mentioned, I think the calculations are probably a little too optimistic and the Phermex line should probably be lowered. Then, though it is penalized by a large factor, the SOC storage ring is much above all the curves--it is a factor of 100 above the improved Columbia synchrocyclotron and Phermex.

Figure 4 shows a comparison of neutron velocity spectrometers in operation, when normalized to the same beam power. The solid line is the Columbia synchrocyclotron, and the dashed line is an electron linac. We can see that, although penalized by a factor of 10 in the production of moderated neutrons and probably by a factor of 2 because of larger moderation time, the Columbia synchrocyclotron is nevertheless a factor of 7 above an electron linac when the power dissipated in the target is the same. The same type of comparison, but for proposed neutron velocity spectrometers, is made in Fig. 5. The improved Columbia synchrocyclotron is above the others. The electron linac is intermediate and the SOC storage ring is below all the others because it is penalized by a factor of 100 compared to the linac. However, it is well above the others when we do not normalize to the power in the target; the power dissipated in the SOC target is quite enormous.

I should like to end with two comments. First, this comparison does not pretend to be very accurate because we lack knowledge about the targets, not only as far as the fast neutron production is concerned, but also as far as the conversion of fast neutrons into moderated neutrons is concerned. For example, we need the relation between efficiency for producing moderated neutrons as a function of the beam power. In the case of a small linac, the efficiency is something like 10% and for other small beam powers the efficiency should decrease very little. For larger powers, however, this efficiency would decrease significantly, but we have little knowledge about this dependence on beam power. We need this relation for proton machines also. The second comment is that I took the comparisons for a definite energy resolution, namely 0.1 times the Doppler width at low energies and 0.5 times the Doppler width at higher energies. If I had taken another criterion, the picture would probably have been quite different because of the different repetition frequencies of the machines.

RAIEVSKI: I have a comment on the number that you have given for the peak power of the SORA reactor. I believe you have given the number of 640 MW at 150 cps, but it is actually 1000 MW for the plutonium core.

KLEY: In your analysis of SORA plus injector, what pulse width did you take and was your conclusion that it is not better than a high-flux reactor with a chopper?

MICHAUDON: I took 5 μ sec. As far as the chopper is concerned, for measurements of total cross sections on small samples, roughly the conclusion is that the high-flux reactor is a little better than SORA plus injector, but with a higher signal-to-

background ratio for the pulsed system.

CEULEMANS: I first want to thank the Chairman for proving that the Nevis synchrocyclotron is still the best machine around,* and secondly, I should like to ask him whether he considers 2000 pulses/sec for the chopper realistic. (*Note added in proof by Michaudon: On the contrary, the Nevis synchrocyclotron was shown to be comparable to the Saclay linac, which is not the best linac now in operation.)

MICHAUDON: I think that this is realistic for the chopper repetition frequency.

BOLLINGER: I would say that if one builds a chopper for this particular class of measurements, it is realistic.

CEULEMANS: Then I agree with the conclusion.

BOLLINGER: In your comparison of the SORA type of reactor and the steady-state reactor, you used a rather large power ratio, whereas in fact it is the flux in which one is interested. I had been under the impression from what I heard a few days ago that the flux available from the pulsed reactors was smaller than that available from the steady-state reactors. Would someone clarify that for me, please?

MICHAUDON: This is the impression I had from discussions, too, but I do not know exactly how much better the continuous reactor is.

RAIEVSKI: No, I would not agree. If you take the moderated neutron flux in the moderator per MW of steady-state power, the pulsed reactor is better because it has a fast core which is very concentrated. The question of Bollinger, I believe, refers to neutrons which can follow the pulse of the linac. If you consider neutrons with energy, let us say, above 1 eV, these neutrons are produced in a pulse of 5- μ sec width. There is no broadening of the pulse in the moderator, and there is no decrease of the peak flux. Thus, in this case, the peak flux in the pulsed reactor is equal to the flux produced by a steady-state reactor of the same type operated at the maximum power.

BOLLINGER: Let me repeat your statement to make sure that I understand it. You are saying that if one had a synchronized chopper looking at such a pulsed source, then effectively the epithermal flux you would be viewing would be substantially higher than that in a high flux reactor?

RAIEVSKI: It would be the same flux that you would get for a 2000-MW steady-state reactor.

MICHAUDON: From Raievski's comment it appears that the ratio of neutron production to power is better with the SORA plus injector by a factor of 2. Thus, instead of being a factor of 2 in favor of the continuous reactor, the counting rates should be equal.

RAJEVSKI: It is not because of the injector. I am taking the case of the thermal, steady-state reactor with a large volume of the order of, let us say, 200 liters. However, the pulsed reactor is a fast reactor and the core volume is of the order of 6 to 10 liters. It is for this reason that you have more n/MW of power, and the figure is about a factor of 2.

BOLLINGER: It seems to me possible that a certain amount of confusion is being introduced here by comparing the under-moderated, pulsed reactor with a highly-moderated, steady-state reactor. After all, one can enhance the flux of epithermal neutrons in the steady-state reactor also, but, generally speaking experimentalists prefer not to do this because it causes background problems from very fast neutrons. Qualitatively, I think I have been answered to my satisfaction--one cannot take the ratio of epithermal neutrons as being the same as that for the thermal neutrons. That is the qualitative information I wanted.

MICHAUDON: Milton will now speak about fission induced by neutrons above thermal energies.

MILTON: I am going to give a very general discussion and I have three or more reasons for doing so. One is that you have already heard the point of view of an experimental fission physicist, and much of what I could say, even though I have been assigned a different energy region, would be just a repetition of what he has said. The second reason is that fission is just a branch of nuclear physics, and all the considerations that apply to doing nuclear physics with steady-state reactors or pulsed reactors apply almost unchanged to fission. We have just heard a very complete review of the characteristics of the various devices from our Chairman. While I do not necessarily agree with what he said, I think there would be no point in just covering the ground all over again. I think that it is very useful to have these things written down so that we can go away and think about them at our leisure.

I am sure you are all aware of the general advantages of higher intensity as given in the following list:

1. Reduced data-taking time
2. Improved resolution
3. Capability for more complex experiments
4. Improved quality
5. Extension of range
6. Small samples or low cross sections

There is nothing very profound about any of these things, and they have been implicit in much of what we have been saying at this conference. Nevertheless, I thought it was worth writing them down explicitly, because I do not believe that it has been done yet. These advantages are things that accrue to nuclear physicists, fission physicists, solid-state physicists, or any others. Let me give you examples of what I mean by

taking my examples from fission physics. The first one is, of course, the very obvious one that one can take data at a much faster rate. As a very primitive example of this sort of thing, the present experiment that we are engaged in at Chalk River, which measures the number of neutrons and their energy spectrum from individual fragments, requires something of the order of three months or more to get reasonable statistics. It would be very much nicer to do this sort of thing in three days. The second item, improved resolution, is one we have heard much about from the solid-state physicists. Improved resolution is something that would be important so that we could do fission experiments from individual resonances. The third item one might do with increased intensity is to increase the complexity of the experiment. Here, fission, I think, is a very good example of the sort of thing that one might do. One might, for instance, want to do a very complicated experiment where one looks at the two fission fragments and the neutrons that came out while an alpha particle was also emitted. The fourth item listed is improved quality of the beam. By that I mean not just improving the resolution, but I mean cleaning up the beam so that the background is reduced and a better experiment can be done. Perhaps a good example of that would be to do an experiment with fission gamma rays. Most of the experiments with fission gamma rays that have been done so far have been done with ^{252}Cf , which fissions spontaneously, because it is very difficult to do experiments with gamma rays around a reactor. It would be nice, even though one does not need very much more intensity, to clean up the beam with some of the methods that Maier-Leibnitz and Armbruster have suggested. The fifth item is the extension of the range of parameters. Fission experiments have been very limited until now by the fact that almost all the experiments have been done with thermal neutrons. If one had a free choice of where one would do experiments with fission, one would not do it with thermal neutrons, because usually one is well above the fission threshold, as Armbruster mentioned. Therefore, one wants to increase the range of fission experiments by having them done at individual resonances. The final item, which to some extent is the same as the first, is that experiments may be done with small samples or samples that have small cross sections. As an example we recently tried to do an experiment on ^{229}Th and found that, with a fairly large fraction of the world's supply that we were able to get, we simply could not do a decent experiment.

I said that fission is just a part of nuclear physics. This means that fission physicists will have to learn about nuclear physics, but, what is even worse, nuclear physicists are going to have to learn something about fission. It was clear at the conference at Gatlinburg (International Conference on Nuclear Physics, 1966) that one of the interesting things now happening is that calculations of nuclear matter are approaching the point where they might actually have something to do with nuclear reality. One of the very interesting things in these calculations is the behavior of the residual forces--how they change with surface-to-volume ratio. The only

experiment which, to my knowledge, has anything to say about this whatsoever is a fission experiment, and it says that the gap which results from the pairing force is twice as big at the fission saddle point, where the distortion is large, as it is at the (deformed) ground state. There are many other examples that could be given.

I should still like to belabor this point that fission is just nuclear physics. One of the things that people, who do fission, would like to know very much right now is the spins and parities of the resonances leading to fission. There are some measurements by Sauter and Bowman in plutonium and very few others. Thus, it would be very nice to have a large body of data on the spins and parities of the resonances. One could then find out what correlations exist.

Another thing that one would certainly like to do is to extend the very nice experiments of John Dabbs on the angular distribution of the fission fragments resulting from aligned nuclei and do these at different resonances. Thereby, the K values, and perhaps the J values, of the total spins of the transition states could be determined. One would also like to look at the relative p-wave and s-wave cross sections of even-even nuclei and determine the parity of the lowest transition state.

Other things that one might like to do are, for instance, photofission. This might seem to be outside the scope of this conference, but some photofission results have been obtained using capture gamma rays by de Carvalho and others. These are very tantalizing, but unfortunately neutron-capture gamma rays come in rather too specific energies to be very useful. Some work has been done by the Russians using a bremsstrahlung spectrum, but I think we are all aware of the limitations of that technique. It would be very nice to have a source of variable-energy, monochromatic gamma rays in the region of 5 MeV, and this is something that one could obtain by the technique pioneered by Knowles at Chalk River. It is just barely possible that fission experiments could be done with a gamma beam obtained in this way if one had a reactor that was about a hundred times more potent than NRU.

We have heard quite a bit at this conference from many people who have said that, while much time, effort, and money are being put into building new sources, we should put a comparable amount of effort into building new instruments. I do not disagree with this, but I should simply note that such things as APR (SORA with injector) and ING are new instruments. They are new techniques, and they may also be new sources, but I think they should be pursued because they do have these new features about them.

Finally, I believe this Panel was supposed to discuss which devices would be most useful for their particular specialty. It was clear to us at Chalk River that the most useful device would be the one that gave us the most flexibility, other things being equal. Of course it has been pointed out at this conference that other things are never equal. Usually the thing that is least equal is the cost. However, in doing fission physics and nuclear physics, I do not think that we are in the same position as the solid-state physicist. We cannot look ahead to an almost unlimited

future of applying well-known techniques, cheering them up a little bit, and doing experiments on more and more substances. The range of energies that the solid-state physicists require is very restricted, because they are concerned mainly with atomic energies, that is, energies of a few eV. Whereas, a fission physicist or a nuclear physicist must cover a very broad range of energies, several MeV at least. Furthermore, each fission and nuclear physics experiment brings with it its own technique. Thus, the experiments that we talk about now, the ones that are listed in all the proposals, are the things that people are interested in doing now. When the machine is built five or more years from now, all the easy experiments in this list will have been done. The harder ones may no longer be interesting to anyone. Therefore, it seemed to us that one should build a very flexible machine, one that was capable of giving very good facilities in both providing a high continuous thermal flux and very good characteristics in the resonance region, the latter could be from either a chopper attached to a steady-state reactor or from selecting individual pulses and moderating the neutrons produced in a manner which could be tailored to each individual experiment.

Like all answers to difficult problems, our answer has many drawbacks. You are all aware, I am sure, that one of these drawbacks is the very high activation in the machinery. Another drawback is the rather frightening complexity of the installation. Finally, of course, there is the cost. Nevertheless, I think that one should be very careful in investing very large amounts of money in a facility that is suitable only for doing, for example, diffraction studies in solid-state physics.

MICHAUDON: I suggest that we defer the comments to the end of the Panel Discussion. Sailor will now discuss polarization and neutron-neutron interactions.

SAILOR: I am going to speak briefly about polarization measurements in the intermediate neutron energy region. For several years there have been methods for polarizing neutrons at thermal energies or in the epi-cadmium range, and there have been methods of polarizing neutrons in the MeV range by various nuclear reactions in nuclear scattering events. It has only been during the last two years that any practical methods have been devised for obtaining strongly polarized neutron beams in the intermediate range from, say, 50 eV up to several hundred keV. The coming of these new methods of polarization has opened several possibilities for experiments in this range, which undoubtedly people will want to be doing in the next few years. I should like to describe, first of all, the characteristics of these methods of polarization and how this affects the type of source that one would want to have to make use of these methods.

The only one that has actually been tested experimentally, and the first one devised, is the method developed at Dubna by Shapiro and his colleagues. The method

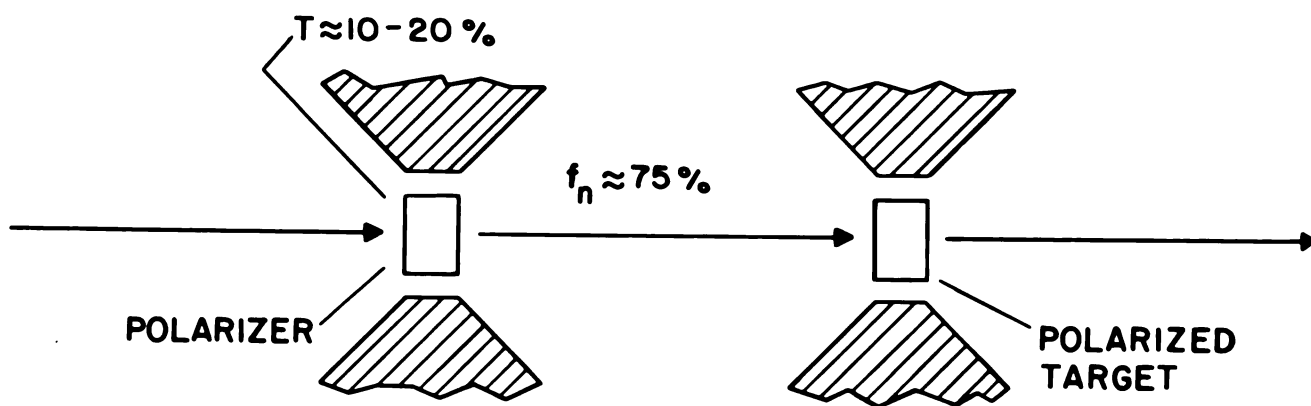


Fig. 6. Neutron polarization by transmission and irradiation of polarized target.

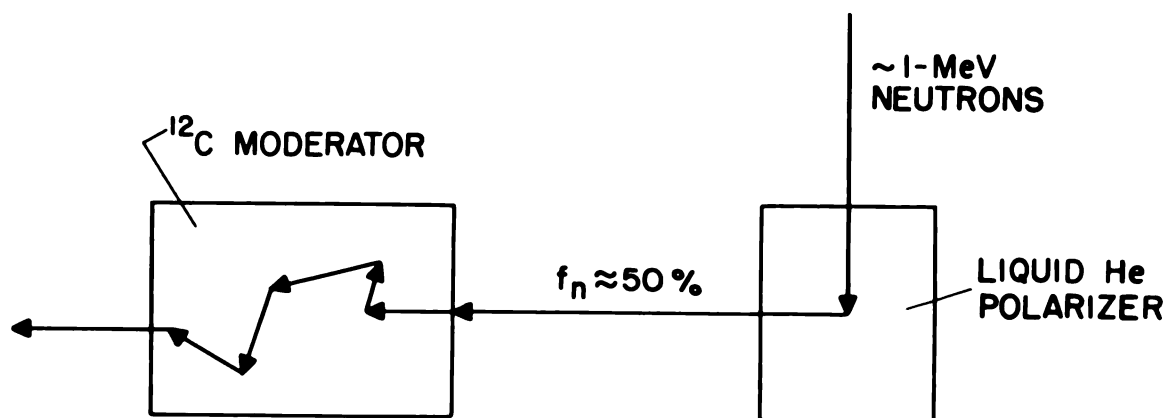


Fig. 7. Neutron polarization by scattering followed by moderation.

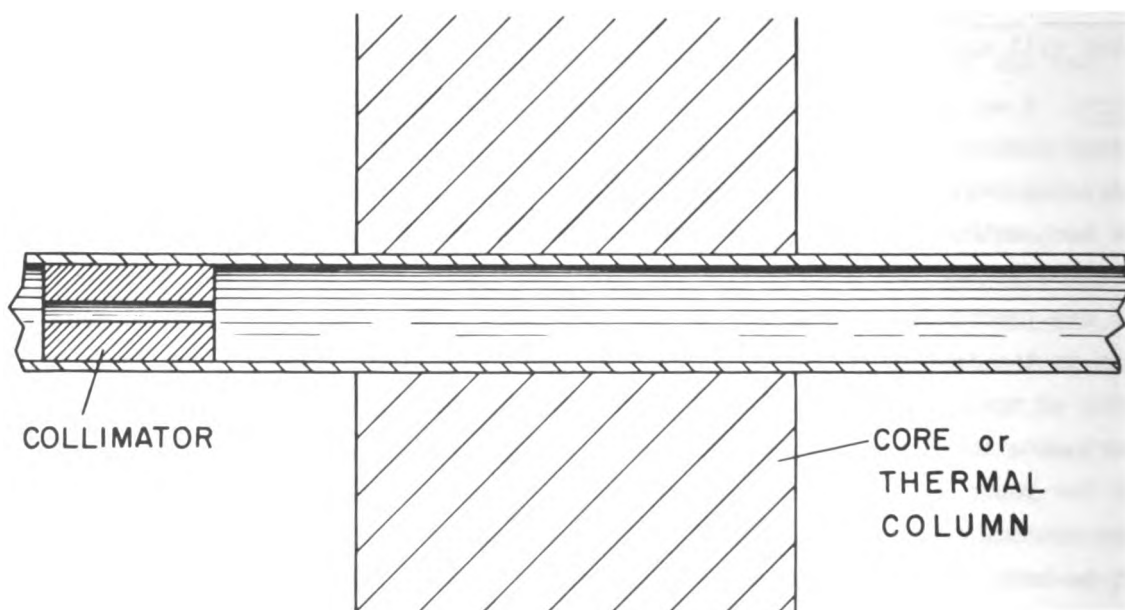


Fig. 8. Schematic of arrangement for measuring n,n scattering.

works on the basis of transmitting the neutron beam through a target containing polarized protons. The proton polarization is done dynamically in a crystal, and it is actually the water of crystallization in this crystal that is polarized. As seen by Fig. 6, the neutron beam passing through the polarizer is selectively attenuated by the polarized protons, so that the neutrons in one spin state are rejected and the neutrons in the other spin state are transmitted. One can obtain then a degree of beam polarization, f_n , which depends on the thickness of the crystal used and on the polarization of the protons. If one uses the present-day technology, one can obtain a polarized neutron beam of around 75% polarization. In this case the beam is attenuated by a factor of 5 to 10.

The polarized neutron beam would then be passed through a polarized target. One could then look at resonances in this target to determine the spins of the resonances or determine the dependence of the strength function on spin. Shapiro has made several measurements with this source, and the results are most impressive. One characteristic of this source is that it is very small in cross section--about 15 cm² at the present time, although I suppose we can expect to see this increase in size considerably over the next few years. The size is limited by the uniformity of field required for the dynamic polarization and by the rf cavity that is required for polarizing the protons.

Any method of polarizing neutrons is inherently very wasteful of neutrons. This is one feature about using polarized neutrons that has to be remembered. First of all, half of the neutrons that are produced in these sources are thrown away because half of the neutrons are in the wrong spin state and all of the methods of polarization depend on rejecting one of the spin states. In addition, with this method of polarization one rejects another factor of 5 to 10 of the beam because of the necessity for the transmission through the polarizer.

The second method which has been developed is the method that Dabbs and Harvey developed at Oak Ridge. This has not yet been tested experimentally, but a great amount of calculation has been done and it looks very hopeful. In this method of polarization, shown in Fig. 7, a neutron of about 1 MeV from an accelerator enters a liquid helium scatterer. As a result of this scattering event, the neutrons will become partially polarized. At a 90° angle, the scattered neutrons have about 50% polarization. The scattered neutrons then pass into a ¹²C moderator. Since there is no spin in the moderator, they can be moderated down to low energies without any appreciable loss of their polarization. Some 14 to 15 scattering events are required to go down each 100 keV or so. It appears that this source of polarized neutrons could be useful in the energy range from roughly 700 keV down to as low as 20 keV. Somewhere in the intermediate energy range, this method and Shapiro's method should become about equivalent. Offhand, it appears that this would occur somewhere in the range of 50 to 100 keV.

The type of experiments which one could foresee immediately for using polarized neutrons consists mainly of programmatic-type work--the measurements of spins of resonances as I mentioned and the determination of the spin dependence of strength functions. We have a little clue as to what kind of intensity is needed to get these data. I think that sooner or later people are going to want this kind of information on essentially most isotopes over the whole range of elements. Judging from the time required for Shapiro to accumulate data to the present time, I expect that another factor of about 100 in intensity would be desirable so that this class of experiment could be done in a reasonable length of time. I suspect that, as this work is done, the demands on precision, etc., as other experiments develop will increase the demand for even more intense sources, by more than a factor of 100. Thus, I think that, whenever new high-intensity neutron sources become available, there will probably be someone there wanting to use polarized neutrons.

Now I should like to comment very briefly on a second point. This is the question of whether or not one should build specialized sources for doing certain types of experiments. I disagree with Milton in his statement that one should not build sources just to do one class of experiment. I think that in the long run this might be the more economical approach to the situation.

In particular, I want to talk about a very specialized class of experiment of the type where you might build a source just to do this one experiment. The example I should like to cite is the case of the neutron-neutron scattering cross section at zero energy. I think most people who have worked with neutrons have looked into the possibility of measuring the neutron-neutron scattering cross section. It does not take very long to convince yourself that this is a very difficult experiment. In fact, until recently it has looked rather hopeless. There have been several serious proposals for measurements. One proposal has been made by Bowman and Dickinson from Lawrence Radiation Laboratory, in which a nuclear detonation would be used. This would measure the scattering cross section in a higher energy range, from roughly 20 keV up to something like 1.7 MeV.

The type of measurement that I am talking about is a measurement at essentially zero energy. Two or three years ago, Carl Muehlhause made such a proposal which looked extremely hopeful and extremely interesting. In fact, I think he intends to go ahead with this measurement at the National Bureau of Standards (see ANL-6797, p.21, and NBS-7583). His method is illustrated schematically in Fig. 8. Essentially, a highly evacuated cavity is placed in the core of a reactor. It is actually a pipe that would run through the reactor from one side to the other, and a collimator is placed in one end of the pipe. In principle, the only neutrons that can come out of this pipe are neutrons which have scattered off each other in the evacuated space if the collimation is correct. When one measures the neutron-neutron scattering cross section by this method the counting intensity that is obtained is

$$I \propto \frac{\varphi^2 R^4}{\bar{v}}, \quad (9)$$

where φ is the absolute flux, R is the radius of the cavity, and \bar{v} is the root-mean-square-velocity. Carl Muehlhaue faces two very serious difficulties. The first is that his flux is rather low for getting measurable results, although it looks like he will get several counts a minute. Perhaps the greatest difficulty is that with such a low flux it is necessary to have an extremely good vacuum in the pipe. It appears that adequate vacuum might prove to be one of the biggest difficulties.

I think the time has now come when it is technologically feasible to make this measurement and make it in the proper way. To make this measurement, a source will have to be built which has a considerably greater flux than is now available to us. One such source has been suggested by J. Chernick, who studied this sort of system several years ago. This is a large, heavy water solution. It is a homogeneous reactor with about 1000 cu. ft. of solution, so that it has a very big volume. One could put a very large scattering pipe through it, thus enhancing R^4 . This reactor would operate in a pulsed mode, being pulsed to a peak flux of possibly as high as 10^{18} n/cm²sec. This pulse would have a duration somewhere in the range of 0.1 sec to 1 sec, and it would have a well-thermalized spectrum. If one were to operate such a reactor in this experiment, it appears that with a peak flux of 10^{17} it would be possible to get around 10^7 n/burst in your detector. Of course, with that kind of intensity this opens up other possibilities such as coincidence experiments. This would look like a very hopeful approach. One possibility for such a source would be the system described by Whittemore of General Atomic earlier in the week. I think it would be very interesting to see if that particular reactor might fulfill the needs for this experiment. I think the time has come when some serious consideration should be given to carrying out such an experiment. Of course, one needs to answer the question: How important are these data and what will be the value of getting this information? I think there is, at the present time, a renewed interest in the question of charge independence of nuclear forces, and it appears that, if one is going to get any useful data, one should aim for something like a 1% precision.

MICHAUDON: Now Bollinger will treat the difficult subject of fast neutron spectrometers.

BOLLINGER: Those of you who know me will be surprised at having me speak on this subject, since you know that I know almost nothing about it. However, this has the advantage that I am at least unbiased, and therefore you can accept the rather definite conclusion that I will make.

As you will see, by highest-energy neutron spectroscopy we are talking here about the range of energies greater than about 100 keV. To stress the special characteristics

of this range, I will start by writing down a relationship you have seen several times already. It is that a meaningful figure of merit M for time-of-flight experiments is

$$M = \frac{Nf}{\tau^2} \quad (10)$$

where N is the number of n /burst, f is the frequency of the bursts, and τ is the time uncertainty. A key point is that one must take into account all time uncertainties, so that

$$\tau^2 = \tau_0^2 + \tau_d^2 + \tau_l^2 \quad (11)$$

a relationship which includes the uncertainty τ_0 of the pulsed source itself, an uncertainty τ_d related to the detector, and an uncertainty τ_l related to the flight path.

In Eq. (10) it is obvious that to have a high figure of merit we want to have a high repetition rate and a very small time uncertainty. Because of the importance of the time uncertainty, we are immediately led, as Havens points out in his paper, to discover that there are two classes of time-of-flight spectrometers: one, the class in which one must moderate the neutrons, and, the second, the class in which one does not moderate the neutrons. If you do not moderate the neutrons, then you may attempt to make the total time uncertainty as small as a nsec. If you accept this statement then you are further led to conclude that the detector, in order to achieve this nsec timing, must be very fast-- like a scintillator. Without belaboring the point, I will ask you to accept it that the kind of spectrometer we are talking about is one in which the pulse of projectiles out of the accelerator is very narrow and in which all the associated equipment is fast enough to permit nsec timing.

We have heard described at this conference several systems that are used for fast neutron time-of-flight spectroscopy. In one of the papers the figures of merit M were given for several of the systems as follows:

<u>System</u>	<u>M</u>
Electron linac (RPI)	3×10^{28}
Synchrocyclotron (Nevis)	3×10^{28}
Isochronous cyclotron (Karlsruhe)	1×10^{31}

The values are for existing systems--the RPI linac, the Nevis synchrocyclotron, and the isochronous cyclotron at Karlsruhe. One sees that the figure of merit is so much higher for the Karlsruhe isochronous cyclotron that we can immediately forget the others for this particular discussion.

All of the machines considered thus far have the characteristic that they have a white spectrum of neutrons. We must now go on to decide whether such a system is competitive with the pulsed electrostatic accelerator, which can have a monoenergetic spectrum, if need be, or at least (in another class of experiments) a spectrum confined to a relatively narrow band of neutron energy. When I first started to

consider this problem, which I was unable to do until I arrived here, it seemed almost impossible to come up with any conclusion because I did not have the necessary literature. Moreover, I suspect that much of the needed information is not in the literature. Then I realized that, from entirely qualitative information, one can draw a valid conclusion. This qualitative information is simply the fact that the total cross section of iron has been measured both with the Karlsruhe cyclotron and by Alan Smith at Argonne with a pulsed Van de Graaff. As it turns out, the results obtained in the two measurements are roughly equivalent. I am not going to be concerned about a possible difference of a factor of 2 or even 5 in intensity--it is unimportant for this discussion. As I remember the two measurements being compared, the resolutions were roughly the same, the counting rates were roughly the same, and most of the characteristics of the measurement were roughly the same--something like a 1-nsec burst width and a 50-m flight path. I do not know what the repetition rates were. On the basis of this purely qualitative information, it seems clear that the fluxes available must have been roughly the same for measurements of approximately equal resolution.

If you accept this, we may now take the next step and ask which machine has better characteristics for doing physics experiments. Consider (Table III) the characteristics of the isochronous cyclotron (white source) and the electrostatic accelerator (reaction source) for various kinds of experiments. The neutron-energy range considered is ~ 0.1 to 2 MeV, a range covered by the ${}^7\text{Li}(p,n)$ reaction for $E_p \lesssim 4$ MeV. The first line of the table compares measurements of the total cross section σ_t for large samples. As I mentioned before, the two accelerator systems

Table III

Comparison of white sources and reaction sources. The symbol + means good and - means poor.

Experiment	White	Reaction
σ_t - large sample	+	+
σ_t - small sample	-	+
(n,p), (n,d) etc.	+	+
(n,t), (n, γ)	+	+
γ spectra	-	+
(n,n)	-	+
(n,n')	--	+
polarization	-	+

are roughly equivalent and both are good for the purpose. The next line is for small samples. Here the electrostatic accelerator is clearly better (+) because all the neutrons are produced in a relatively small spot. Measurements such as (n,p), (n,d), and (n, α) are the next class considered. Whether or not measurements of

this sort can be done well presumably depends entirely on the flux that is available. Thus, here again, the two systems are equivalent. For the same reasons, fission and capture gamma-ray cross-section measurements could be made with approximately equivalent ease with the two systems, insofar as counting rate is concerned. However, gamma-ray spectra of good quality probably could not be made with the white source. This is because the neutrons scattered into the gamma-ray spectrometer generate background gamma rays that would be a problem, whereas with the pulsed electrostatic accelerator one can eliminate this problem by a pulsed technique that many of you know of. For elastic scattering of neutrons, again the electrostatic accelerator is clearly advantageous because of the feasibility of using the time-of-flight technique with monoenergetic neutrons to make sure that elastic scattering is being observed. This statement is even stronger for inelastic scattering. For polarization, once more the electrostatic accelerator is to be preferred because there are good reactions that produce polarized neutrons, whereas even with the Dabbs and Harvey method of polarization, which looks promising for a white spectrum, there is an intensity loss of a factor of 200 or thereabouts.

Before arriving at a conclusion, which might be clear by now, I should note that I have only mentioned one class of measurements that can be made with the electrostatic accelerator, the class that uses a pulsed beam. There is, in addition, the class that uses a steady beam, and this class of measurements is advantageous for certain types of experiments. I am, therefore, compelled to state that, if one considers existing systems, there is no question whatever that one would have to choose the electrostatic accelerator as being far better than the isochronous cyclotron for measurements in the neutron-energy range 0.1-2 MeV. Which accelerator one would choose for the future is not quite so clear, as is true of all such questions. However, I feel one would arrive at the same answer because there appears to be good reason to believe that both types of systems will be improved substantially in the next few years.

When we go above the range of 2 MeV, one would use the same kinds of considerations. I have not done this because I have neither the data nor the understanding of the data that would allow me to do it well, and also because this is a topic that Bartholomew will discuss. My own guess, and it is not much more than a guess, is that the electrostatic accelerator would still be superior for neutron energies up to about 25 MeV, a limit set by the $T(d,n)$ reaction used in conjunction with the kinds of high current electrostatic accelerators that are now available.

MICHAUDON: I suppose there will be several comments.

HECKKURTS: I am not prepared for this very detailed analysis, so it may take me some time to present the arguments. First, I do not think we have seen these data of Alan Smith. I think those which have been published were not able to resolve these iron resonances. They must be fairly recent data. A second point is that we believe

the resolution of our spectrometer is not what it might be. Those data, which we have taken on iron and have shown, were very early results that were not taken with full capacity of the machine. I think with the machine at its full capacity such data could be taken in a very short time, much shorter than with a Van de Graaff accelerator. Next, I should say that one advantage of the white source method is that with one run you get all the data from about 400 keV to about 20 MeV, which is normally not possible with an electrostatic accelerator. The white source especially supplies neutrons in the energy range between 7 and 11 MeV, where it is a little bit difficult to get neutrons from (p,n) or (d,n) reactions. This region is, of course, now partly bridged by using the $^9\text{Be}(\alpha, n)$ reaction; nevertheless, it is a little inconvenient to work with that method.

As far as the cost argument is concerned, I should like to point out that this is just one part-time use of the machine. I should completely agree with you that it would not be worthwhile to use such a machine entirely for such experiments. As far as the various classes of experiments are concerned, I agree with you that there are types of experiments which cannot be done very well with such machines, especially small sample experiments, which certainly can be done better with a Van de Graaff machine, as we explained earlier.

BOLLINGER: There are a number of points to which I should like to respond. First of all, I hope the audience does not feel this was an attack. I have the greatest admiration for what has been done with the Karlsruhe machine. I am merely attempting to make a clear-cut choice of what is the best accelerator for the purpose. Secondly, as far as I am concerned, measuring the total cross section with slightly better resolution or even considerably better resolution is of almost no interest. What one really wants to do is to make a variety of measurements of the kind that appear at the bottom of Table III; these are the scattering measurements and measurements of various reactions. Therefore, the fact that one might push the technique with the isochromous cyclotron farther to get better and better resolution is of relatively little interest unless one can also do something about measuring inelastic scattering, elastic scattering, polarization, and the rest. That perhaps covers the main points.

TASCHEK: I might make a couple of points here, one of which is that there seems to be a comparison between a white source and a monoenergetic source. I think you all recognize that an electrostatic accelerator, let us say a Van de Graaff, can make a white source and has indeed been used this way when it seemed desirable. In fact, one can make a better white source than the kind that we have heard about on the Karlsruhe machine, although it is not excluded on the Karlsruhe machine either. Namely, one can make a segment of a white source. By using a reaction with the appropriate thickness target, one can, if desired, limit the upper energy and the lower energy. This is often an advantage. The other point that should be remembered

is that both of these machines are charged-particle accelerators. The thing that gets more neutrons is more intensity. In a way, the thing that gets better resolution is to trade intensity for resolution when you need to do that. This can, of course, be done on the electrostatic accelerators when required, namely, by taking smaller acceptance angles from point sources you improve resolutions or you can improve resolution with time of flight. A more important point, which is only now beginning to be overcome with the electrostatic machines, is that in the past you could not always use time of flight in an optimum way by using every bit of the beam. Now you can do that throughout the full range by using klystron bunching, which is coming into use at Aldermaston, Oak Ridge, and elsewhere.

BECKURTS: I should like to comment on the reply of Bollinger. First, it is not only total cross sections which we intend to measure, but also (n,p) and (n,α) measurements. In fact, this can be done with the white source by the use of thin foils. We have such experiments in progress. Another interesting type of experiment, for instance, is the measurement of the fission mass distribution as a function of energy. We are also thinking about some inelastic scattering measurements using double time-of-flight methods, but these are not very well established at this moment. Thus, I think our program includes a really very broad class of experiments in addition to total cross-section results. Also, we believe that even the total cross-section results with a high resolution are very interesting in view of the intermediate-structure interpretation.

BOLLINGER: As I mentioned in my first reply, I also find the (n,p) , (n,d) , and (n,α) class of experiments very interesting. However, the thing I question is whether, in the range below 2 MeV (which is principally the range about which I was talking) one has enough flux with any source to make more than a few rather specialized measurements. I personally do not know enough about it to offer more than a guess that the coulomb barrier will make it extremely difficult to turn this kind of study into a major program until we have substantially more flux than is available now. (Editor's note: Further remarks on this subject by Bollinger appear in the next Panel.)

RAJEVSKI: I would like to come back to the comparisons that Michaudon has made between a steady-state reactor of 100-MW and a 1-MW pulsed reactor with injector. You have taken a repetition rate of 2000/sec for the chopper, which would be used with a steady-state reactor. You have taken 150/sec for the pulsed reactor with injector, so the ratio of repetition rates gives an advantage of about 14 for the steady-state reactor. Now, the power ratio for a 1000-MW pulsed reactor plus injector to a 100-MW steady-state reactor gives an advantage factor of 10 for the pulsed reactor plus injector. With these opposing ratios, you arrive at approximately equivalent devices. I do not argue about a factor of 2, but for the same power you would get about three times more epithermal neutrons with a pulsed reactor

using a fast core than with thermal reactor in steady-state operation. Thus, I believe that you should slightly change your statement and say that the devices are equivalent from the point of view of counting rate.

MICHAUDON: Yes, I agree with these comments that there is a factor of 2 in favor of the pulsed reactor with an injector, but, since I do not want to be concerned with a factor of 2, let us say that they are roughly equivalent. However, there is an advantage from background considerations for the pulsed reactor with an injector.

Do you agree with such a statement?

RAJEVSKI: I am satisfied with that.

MICHAUDON: If there are no other comments, I shall briefly summarize with Fig. 9 which is a rough comparison of the interest in the various neutron sources as a function of the energy. For thermal neutrons, we are interested in such a source for fission physics, and a high-flux reactor or a thermal source like SOC is very interesting. If we increase energy slightly to resonance neutrons, we find the pulsed reactor, and then with an onset around 1 eV--I should not quarrel about a factor of 2--we find the boosters (including accelerator-pulsed reactors). Thus, the booster is superior to the linac and the synchrocyclotrons up to around 100 eV. Now above a few hundred eV, because of source considerations, we find the linac and synchrocyclotrons with possibly an energy of 600 MeV. I think the future of the linacs is quite good. However, because of their ability of target heat dissipation, probably in the long run the proton synchrocyclotron would be superior. Then above a few hundred keV, Bollinger has shown that electrostatic accelerators are probably superior up to 2 MeV. Above 2 MeV is still an open question. Probably the sector-focused cyclotron or an electrostatic accelerator are superior. Just now I don't know, but I leave it to your thoughts.

Again in Fig. 9, if you consider small samples, then up to a few hundred eV we can consider a chopper on a high-flux reactor or an accelerator-pulsed reactor. A nuclear explosion is outstanding for certain classes of experiments from 20 eV, which is a cutoff. And now the SOC with a storage ring, when necessary, if it works--and I hope it will--is certainly an outstanding source which is superior to all the others.

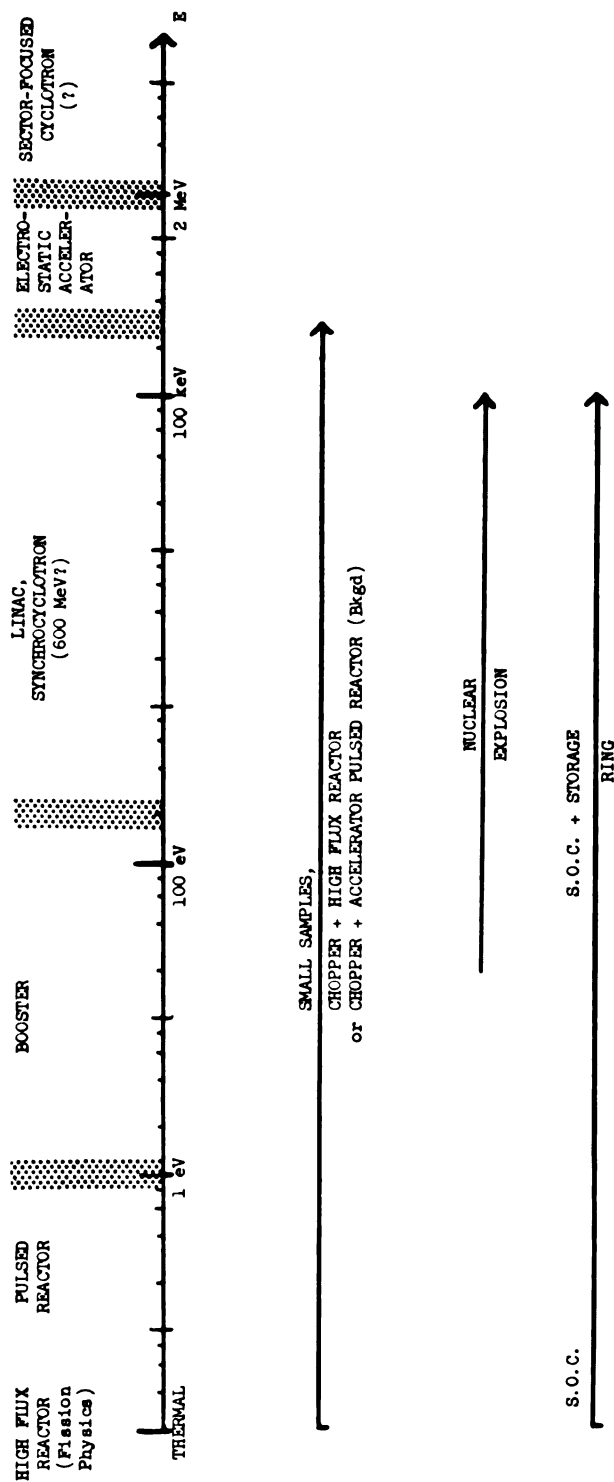


Fig. 9. Summary of types of neutron sources and their applicable energy ranges.

SESSION VI PART D - PANEL DISCUSSION

ON

NUCLEAR PHYSICS

Panel Members: G. Bartholomew, Chairman
H. Maier-Leibnitz
R. Ringo
O. Schult
J. Vervier

Secretary: M. Bunker

BARTHOLOMEW: The object of this Panel is to define the required properties of a high flux source for various classes of experiments in nuclear physics, and attempt to select the best high flux source for each application. We have reduced our considerations to six topics:

1. Neutron Fundamental Experiments
2. (n,γ) Spectroscopy with Thermal Neutrons
3. Rare Isotope Production for Decay Studies
4. Production of γ -ray Sources by Thermal Capture
5. Resonance Neutron Capture γ Rays
6. Fast Neutron Nuclear Physics

These are by no means supposed to be exhaustive, but, in view of the limited time and the fact that many of the things we can say about nuclear physics have already been said by earlier Panels, particularly the last one, we shall restrict our discussions to these. Collectively, the Panel members have had little or no direct experience with some of these topics, and we shall be calling on the participants for generous help in some areas.

For many classes of experiments the neutron only serves as a means of acquiring a nuclear excitation; the experiment does not really depend on the neutron per se. One can imagine exciting the nucleus by other reactions and doing exactly the same experiment, although this might not be quite as convenient. For excitation experiments, requirements on the particular properties of the neutron beam may be somewhat relaxed as compared to those needed in cross-section measurements, for example, which were discussed in the previous Panel. So in some ways our task is easier, and there have been few if any arguments among members of the Panel concerning beam requirements. Let me point out, in passing, that, as a means of exciting the nucleus, neutrons have certain advantages. There is no coulomb barrier problem and neutrons excite the nucleus to quite a high excitation energy. One can excite nuclei throughout the mass table with comparative ease, etc.

Now, the Panel will consider each of the topics in turn and then we will open the discussion to the floor.

The first topic is Neutron Fundamental Experiments. We have already had some earlier discussion of one of these experiments, neutron-neutron scattering, by Sailor; but there are others, and I will now turn over the discussion to Ringo to make some remarks on this.

RINGO: The subject of fundamental experiments on simple systems, primarily neutrons, has already been very well reviewed by Bollinger. I have practically nothing to add to the general physics of the situation except to go into a little more detail on source requirements. But, before I go into this detail, I would like to make a short, heartfelt preliminary comment, which is by far the most important thing I want to say. This comment has to do with my opinion that many people, here and elsewhere, have the impression that there is only a weak enthusiasm among the physicists for high-flux neutron sources. I would like to reply to this for at least a small aging band of us who feel very differently. We think that there is very much point to much higher neutron fluxes, and perhaps the reason for this can best be illustrated by the chance of finding a time-reversal failure in one of these neutron experiments, particularly experiments with polarized neutrons. I think this might be the most important discovery that could be made in physics in this decade. The importance of a high flux to these experiments is quite simple. What one would probably find, based on evidence from the K-meson decay experiments, is a time-reversal failure of say, $1 \pm 0.2\%$. This would be a real sensation. If, however, you find a time-reversal failure of $1 \pm 1\%$, your paper probably will not even get published. This factor of 5 in experimental error, which represents a factor of perhaps 25 in flux, simply means that the experiment could perhaps be done at a high flux reactor but not done at an ordinary flux reactor. Put another way, you could do the experiment in one year at a high flux reactor, whereas it would take some ridiculously long time like 25 years to do it at a low flux reactor.

With this major point made, let me get down to some of the details of the other experiments. The time-reversal experiment, after all, is only one of several, and indeed will probably not work; in other words, time probably will reverse. This is a very heterogeneous group of experiments, but fortunately they have a certain amount in common as far as sources are concerned. Virtually all of them depend directly on average flux and are not particularly sensitive to pulsing, that is to say, they do not demand pulsing. On the other hand, if a source happens to be pulsed, this is all right, although very short pulses such as those produced by explosive devices may indeed cause serious complications. But, roughly speaking, all of these experiments are dependent upon average flux over a period, say, of a year--I mention a year because these experiments are the kind that one might consider spending a year on, or some such order of magnitude. Other characteristics of the source are of course

important, such as background. However, with modern means of handling background, such as single-crystal filters, background is probably not terribly critical. The source area is also important, but once you have an area of, say, more than 10 cm^2 , it is again probably not very critical. So what really counts for the first group of experiments I want to talk about is the average flux divided by the square root of the temperature, because, in general, the colder the neutrons, the better. These experiments include: (1) the neutron lifetime experiment, which is very much a flux-limited experiment; (2) asymmetry in the decay of polarized neutrons, which again is very much a flux-limited experiment; (3) the tests of time-reversal symmetry, as I have already mentioned, including the one involving the decay of polarized neutrons, which is certainly a flux-limited experiment. The time-reversal test done by capture in such nuclei as ^{48}Ti , with study of the subsequent gamma cascade is perhaps less flux-limited with the existing techniques, but there is always a hope that the detector systems can be improved, shorter resolving times be attained, etc., so that one could go to higher fluxes and would benefit by higher fluxes. Next, the neutron electric-dipole moment measurement discussed in considerable detail by Bollinger is to a certain extent a flux-limited experiment. Then there are even stranger experiments that might get interesting if they could be done to much higher accuracy; e.g., the gravitational constant measured with neutrons is a flux-limited experiment if you want higher accuracy. The neutron-proton interaction experiments, in general, are perhaps not flux-limited, but it is very hard to be sure exactly what kind of technology one is going to use with these experiments in five years; it seems reasonable that some of the things one might want to do would be greatly benefited by higher flux.

There is a second group of experiments that do not utilize cold neutrons, so their figure of merit is simply the average thermal flux. These include experiments such as study of the electron-neutron interaction, which can be done by several methods. The one I am most familiar with--the scattering in noble gases looking at the asymmetry of neutron scattering by the electron cloud--is flux-limited. The parity-failure experiments in cadmium, the Aboff experiments (which were invented by Adair), are not flux-limited at the moment, but, again, changes in the detector systems might make them so. Finally, the neutron charge experiments are to a considerable extent flux-limited.

I think this covers the bulk of the immediately interesting fundamental neutron experiments. There are several other experiments of this general character on which Maier-Leibnitz might like to comment.

MAIER-LEIBNITZ: One further experiment is the so-called neutrino experiment. At the last Geneva Conference, the Russians suggested building a relatively large reactor and letting a big fuel element fall down into it, so that the whole thing heats up to 2000°C . The reactor then stops by negative temperature coefficient, and of course there is a lot of radioactivity afterwards, which can be used to detect

neutrino effects. I am not sure how favorably this method compares with other possibilities, but I think it must be considered. Another use of the same arrangement might be to try the experiment Sailor has spoken about, the neutron-neutron interaction. In this case, one should be able to provide a rather large beam hole, so I think it should be quite favorable. Again, I do not know all of the details.

Now, I might mention a further small thing. Using our gravity refractometer, we can measure coherent cross sections to within 10^{-4} , which may have theoretical interest in some cases, but an especially interesting application is to measure the neutron-electron interaction by the old method of Fermi and Hughes. We also have an interferometer which could, in principle, be used to measure the electric-dipole moment of the neutron. We could build a device to flip the spin so that one path has neutron spin up and the other has neutron spin down. Then they are made to travel through inhomogeneous fields, which turns them around all the time, and in this way you can build sort of an accelerator based on the dipole moment of the neutron. I think this should be superior to the methods so far proposed.

Finally, I should mention that with the back-scattering method, which seems to be quite accurate if you try very hard and have enough flux, you should be able to go beyond, say, 10^{-4} in resolution in double back-scattering of neutrons. Now, if one of the crystals is under ultrasonic vibration so that the surface vibrates with a certain velocity as a function of time, then the neutrons will not be reflected from the second crystal unless it is vibrating with the same velocity and phase as the first. In any case, using this resolution, you have a very accurate neutron energy which can, in turn, enable a very accurate measurement of velocity. What you finally measure is h/M in units of velocity and the X unit (the lattice distance in a crystal). I should mention that this idea of ultrasonic vibration is not ours, but that of Stepman at Studsvik.

BARTHOLOMEW: The second item on our list is Neutron Capture Gamma Rays, and here we include simple spectroscopy of the one-parameter type, i.e., measurement of energies and intensities, including internal-conversion spectroscopy, and also we will include multiparameter spectroscopy, that is, coincidence measurements, polarization measurements, circular polarization measurements, etc. First, I will ask Schult if he would briefly summarize the one-parameter type experiments.

SCHULT: Much information on nuclear structure can be gained through measurements of gamma- and conversion-electron spectra from neutron capture. Lithium-drifted germanium diodes, magnetic spectrometers for the detection of electrons, and curved crystal spectrometers [predominantly for the low energy (n,γ) region] are commonly used for this purpose. The last two devices require very small sources, and therefore high specific activity of the source. It is easy to show that a very small source is also of advantage during measurements with germanium spectrometers.

Therefore, all three methods use a geometry where the source is usually located in tangential through-holes situated close to the reactor core. The quality of data which can be obtained is given through what we may call sensitivity, and this sensitivity has to be considered for the whole device, namely, the instrument and the reactor with its neutron and gamma-background spectrum. If the instrument outside the reactor is shielded sufficiently well, then the sensitivity is mainly given by the ratio of the thermal neutron flux to roughly the square root of the gamma flux at the source position. For a fixed geometry, the sensitivity will increase approximately with the square root of the neutron flux. So here we do not gain as much as we would hope. In answer to the comment by Shaftman, namely, that the experimenters should specify their requirements a little more clearly, we feel that a considerable improvement of the sensitivity would be obtained if reactor designers could succeed in building reactors which have a reduced gamma background at the source position in the tangential beam tubes. Apart from this, the sensitivity can be maximized by choosing a favorable distance between these beam tubes and the reactor core. According to the sensitivity as I have defined it, this should be very simple.

Now, I want to show you a figure (Fig. 1) from which we can learn a little about what can be accomplished with good reactors existing at the present time, and we will also learn how much more flux would be useful. In this figure I have plotted the density of isotopes in a certain cross-section region. At the extreme right we have a Mb, and at the extreme left a tenth of a mb. At existing 10-MW reactors of the types DR-3, at Risø, or CP-5, at Argonne, reasonably good bent-crystal spectra can be obtained if the target isotopes have a capture cross section of $\sigma \gtrsim 30$ b (indicated by the region to the right of the dashed line). Given a sufficiently large target mass, germanium-diode spectrometers can be used to study nuclei with somewhat smaller cross sections. Now, you can see from the graph that a reduction of the 30-b limit by a factor of about 300 would make low-energy (n, γ) experiments feasible for the majority of available isotopes. Since we now have a flux of about 6×10^{13} n/cm²sec at the target position, the flux needed to achieve the desired sensitivity is about 2×10^{16} n/cm²sec. Of course, it has been tacitly assumed here that the people doing the experiment will at the same time optimize and extend their equipment to an extent comparable with the amount of effort required to build the higher-flux, steady-state reactor.

BARTHOLOMEW: I will just say a word or two about other types of capture gamma-ray experiments with thermal neutrons, such as the gamma-gamma coincidence or angular correlation experiments, and some of the others that are more complicated. These can perhaps be divided into two classes, but they are all beam experiments. In coincidence experiments, one prefers to bring a beam of thermal neutrons out of the reactor and set up the experiment outside. The data collection is always slow, so

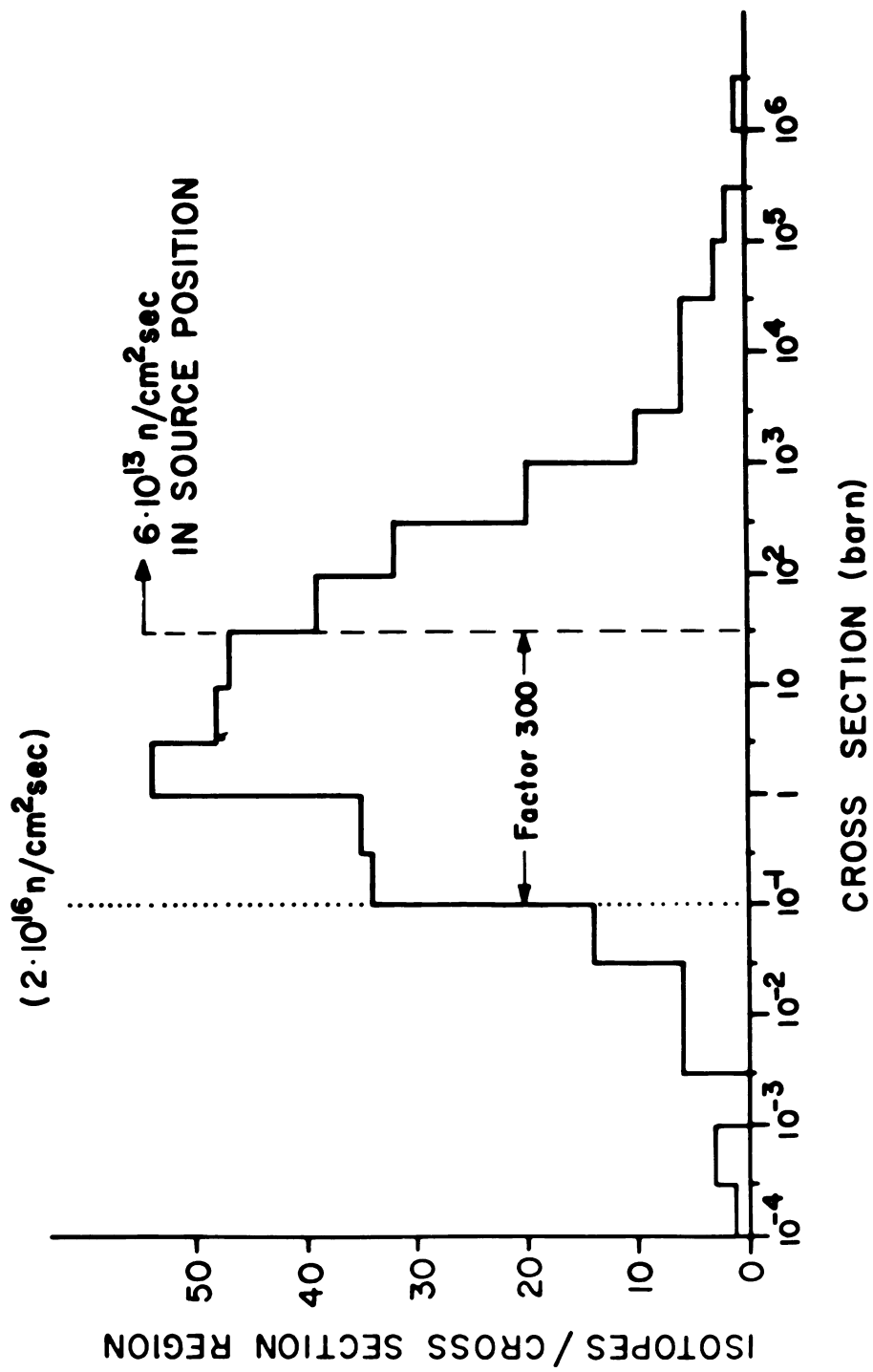


Fig. 1

one tries to maximize counting rates to the extent allowed by the overload of single rates in any one detector and by other limitations, such as random coincidences. One can also say that for such experiments you usually require a rather small source, that is, a small target, on the order of 3 cm^2 in area. Therefore, if your detectors have a solid-angle \times efficiency product of about 1% , and you want to count, say, no faster than $10^4/\text{sec}$ in any one detector, which is around the limit we usually assume, then the maximum source intensity that you can tolerate is of the order of 10^6 captures per sec. This, then, definitely limits you to the cross section times the number of atoms that you can use in that target. At the present time, I think we are limited to sources of the order of 0.1 to 0.01 atom-barn. Obviously, one would like to be able to do other targets, and I think one could stand an increase in flux of a factor of 100 or so; but, at the same time, it is most important to improve the geometry of the neutron beam tube. The beam can conceivably be cleaned up so that this increased flux does not also raise the background. One can imagine, of course, that the beam would strike other parts of the experimental hole and raise the background, or the room background may be high, so that such experiments would only be feasible if they are accompanied by an improvement in the design of the apparatus to go along with it--a factor which has been mentioned several times at this conference.

The other class of experiments in this area is one that is just barely possible today, namely, the measurement of multipolarities of high-energy gamma rays, using the angular correlation of the electron-positron pair from internal pair production. In this case, you are limited by the thickness of the source (or target) that you can put in the neutron beam--the target has to be thin enough to allow the electrons to get out and still preserve their angular correlation. Here again, I think that one could certainly stand increases in flux of the order of 100 or more, including the amount you would need to clean up the beam.

VERVIER: In addition to the methods just mentioned for determining spins in neutron-capture, gamma-ray decay schemes, I would like to mention another experiment that should be interesting and can only be utilized to its full capability with very intense neutron sources. These are measurements of the circular polarization of the gamma rays emitted after the capture of polarized thermal neutrons. Such experiments were performed for the first time by Trumpy, in Norway, and then later by groups at Chalk River and in Czechoslovakia. They all made use of sodium-iodide detectors. The field of capability of such measurements has been severely limited by the energy resolution of these gamma-ray spectrometers. Such measurements should be reconsidered with the large germanium detectors now available. However, a very large flux of polarized neutrons is needed since the efficiency of the germanium detector is rather small. It should also be pointed out that the effect to be measured is very small (the change of counting rates in the spectrometer when the spin direction of the polarized neutrons is changed is of the order of a fraction of a percent), so

that such experiments should only be attempted with large Ge(Li) detectors and with very high fluxes. These experiments would be very useful because the information obtained on the spins of the levels populated by neutron-capture gamma rays would complement the indication obtained from the gamma-gamma angular-correlation measurements just referred to, which are often rather ambiguous. Furthermore, since one is not limited by the resolving time of any coincidence circuit, one can go far enough from the source so as not to be troubled by the counting-rate problem; consequently, an increase of flux of about 10 to 100 would be extremely useful.

BARTHOLOMEW: The next topic is Rare Isotope Production for Decay Studies, and here again Vervier is going to say something about this.

VERVIER: The study of the decay schemes of short half-life isotopes or isomers could be considerably extended with high flux reactors. Here, one can divide the range of half-life roughly in two parts. For half-life larger than about 1/10 sec, a convenient method is to use a fast rabbit. The fluxes now available in existing reactors are, I think, satisfactory for studying the gamma-ray spectra from the decay of such isotopes, even with such a relatively low-efficiency spectrometer as the germanium diode. However, for studying internal-conversion electrons from these decays, high fluxes are obviously needed since the source area and thickness are limited by the energy resolution to be achieved in the electron spectrometer. Such measurements would require the fast introduction of the sample to be studied into the vacuum chamber of the beta spectrometer, and this could be achieved by a fast vacuum lock or, alternatively, one could use a vertical tube traversing the reactor, in which the vacuum would be achieved. The sample would be introduced at the top of the reactor, allowed to fall down by its own weight in the vacuum, stopped if needed at the irradiation position in the center of the reactor core, and then allowed to fall again under gravity, to finally stop at the source position of the beta spectrometer. I think that such a scheme would be very useful for isotopes with a half-life from about 1/10 sec to a few minutes, and this would of course require the reactor design to provide a vertical through-tube for such an experiment. For a lifetime smaller than 0.1 sec, I think that the pulsed reactor would be advantageous. For instance, with the SORA reactor, which has a burst-width in the 50- μ sec range with pulsed intervals of about 20 msec, one could conveniently study isotopes and isomers with half-lives from a few μ sec to about 40 msec using this method. Then the sample could be placed in a tangential hole of the reactor, and the gamma- and conversion-electron spectra could be measured with a germanium detector and a beta spectrometer in much the same manner as was just described by Schult. I should add that with the TRIGA multiple-pulsed reactor, which has been talked about this week, the same technique could be applied with a smaller range of lifetime, perhaps from 1 to \sim 40 msec.

SCHULT: I would like to add a few words to what has just been said by Vervier. It is

true that pulsed devices, and even a rabbit, could be used for the study of lifetimes and the decay of isomeric states, but this can also be done in a different manner, namely, by using a tandem accelerator, as we heard earlier today. These machines are frequently utilized to accelerate deuterons for (d,p) reaction studies. In this process many nuclear levels are excited, which decay to various lower-lying states, including isomeric levels. The heat generated in the (d,p) targets requires the use of small samples, and these small samples can be directly placed in a conversion-electron spectrometer, which serves for the determination of the energies and multipolarities of the transitions involved. This method has been applied very successfully by the Copenhagen group at the tandem in Risø, where the deuteron beam is pulsed and where the actual measurements are performed in the time interval between the deuteron pulses. This technique can also be applied during the study of other kinds of reactions, for instance, (d,t). Through such pickup reactions one can excite levels in nuclei which cannot be reached in the neutron capture process. Therefore, I feel that the study of isomeric levels in the range from, say, minutes to fractions of microseconds, or even to a few nanoseconds, could be done at accelerators a little more successfully and easier than at a reactor.

VERVIER: In the method proposed by Schult, it should be mentioned that some of these isomers can probably not be populated in the (d,p) reaction because they correspond to excited proton configurations, whereas they could be excited by neutron capture; so maybe in some cases the two methods can be complementary.

SCHULT: I do not know whether I should agree completely. One excites very high-lying levels in the (d,p) process, and these levels usually have very complicated structures. When these states decay, lower-lying levels can be populated which are not directly populated in the (d,p) process, and some of these might be proton excited states.

BARTHOLOMEW: I think we should take up the next topic, which is the question of the production of gamma-ray sources by thermal neutron capture, the gamma-ray sources then being used for gamma-ray resonance excitation experiments. Maier-Leibnitz will say something on this subject.

MAIER-LEIBNITZ: Actually, I think I will make it a little more general. In these experiments, you would produce gamma rays either within the reactor core or near the reactor core in the through-tube. In this way you get line spectra, and of course such lines mean that in comparison to any bremsstrahlung spectrum, the phase-space density of those lines is almost infinitely larger. Now, what use can we make of this? The main use that has been made is to try to resonance-scatter these lines from any isotopes with unknown resonance lines, and actually Fleischman and Ben David, in Israel, have found quite a number of such cases. One can then study these lines for their widths, which are mainly Doppler widths, for their strength, and for their spin. One modification I should mention is: One might look not only at the thermal

spectrum but one might also try to go beyond the thermal spectrum if there are still gamma captures. Here, of course, the gamma wave length (or energy) must shift along with the neutron energy. This has not been found so far, but with a reactor of higher flux and with careful consideration of background, I think this should be possible. A high-flux reactor would definitely widen the range where one can make good measurements of this kind. One can also do (γ, n) and (γ, f) reactions. I think this has been done, but I am not too familiar with how useful this can be. Again, you might aim at very accurate measurements.

Another thing you might try, and this has also been done, is to degrade the monochromatic gamma rays in Compton scattering and so obtain nearly monochromatic gamma rays of varying energy. I understand Delbrück scattering has also been found, and the results are being discussed now. I assume this work will be continued. Then recently, in quite a number of laboratories, low energy gamma rays from neutron capture have been used as Mössbauer sources. This, again, has quite a number of applications. I should also mention that from internal conversion you get collimated beams of electrons up to 7 or 8 MeV. In Munich, we have found internal conversion up to those energies, so you have very monochromatic and well collimated electron lines of this kind available for test purposes. The last thing--I do not know whether it should be called nuclear physics--is that recently we have found optical light from nuclear reactions, namely, from beta decay. I believe this is the first time this has been done. If you transform krypton by beta decay into rubidium, what you have is a singly-ionized rubidium atom, and optical lines will be emitted because you suddenly changed the charge. The energy levels that have been excited all have spin zero and angular momentum zero; that is how it must be because the force is central and does not change the angular momentum, and the spin cannot be switched because it is electrostatic. This may be the beginning of a field which could lead to the study of such things as the hyperfine structure of isolated atoms in solids, the electron-neutrino correlation for forbidden transitions in a gas by measuring the Doppler effect, etc. First, one must succeed in making a high resolution spectrometer with an acceptance angle of, say $1/10$ of a radian, and perhaps use it in a coincidence arrangement. I wanted to mention it because it may develop in the future, and it will require quite a lot of flux if one wants to do it well.

BARTHOLOMEW: Can you say something about the flux needed--or does one just want unlimited flux for these various experiments?

MAIER-LEIBNITZ: Well, that depends. You can do such experiments with present fluxes, and you can do better experiments at higher flux. I do not know what the limit is; I suppose something like 10^{32} !

BARTHOLOMEW: The next item is Resonance Neutron Capture Gamma Rays, and here, of course, one measures gamma-ray energies and intensities, and one would ultimately like to

do multiparameter experiments just as we now do with thermal-neutron capture. Some of the best work that has been going on lately in this field has been done at Saclay, and Julien has prepared some remarks which Vervier will read.

JULIEN (read by Vervier): "During these days, we have heard many comments about different solutions of achieving intense neutron sources. Various experiments requiring high fluxes have been described, but orders of magnitude have rarely been given. I shall attempt to outline the specific requirements of a particular field, namely, neutron capture gamma-ray experiments by the time-of-flight method. These experiments give much information on decay schemes and the fluctuations of transition intensities from resonance to resonance. Up until recently, people used NaI(Tl) crystals. Even when computers were used, the results were poor. Even if the neutron fluxes had been 10 times higher, the quality of the data would have been about the same. Now, Ge(Li) detectors are available and, in spite of their low efficiency, the philosophy of the experiments is drastically different.

Our aim in such experiments is to get the maximum number of capture events. To do this, one can increase the neutron flux, increase the weight of the sample, or use a shorter flight path. To give orders of magnitude, with a flight path 28-m long, a pulse repetition rate of 500/sec, and a pulse width of 100 nsec, two or three weeks of operation were necessary to study the gamma-ray spectrum of 30 levels in ^{195}Pt , the neutron energy range being from zero to 700 eV. The data processing is simple, and the information obtained is very reliable. The same experiment carried out two years ago with an annular sodium-iodide crystal associated with a central crystal required, under the same conditions of operation, six or seven weeks, and the data processing was long and tedious in spite of a good knowledge of the detector response.

Because of the low efficiency of the Ge(Li) detectors, we use very large samples. In the platinum experiments, 8 kg of natural platinum were used, i.e., 3 kg of ^{195}Pt . The intensities of the weakest lines studied were roughly 1%. Of course, if only a few tenths of a gram of separated isotope had been available, the neutron flux would have had to be 50 times higher than the actual Saclay linac flux. For small samples, there is also the problem of solid angle with small detectors. One can hope that large Ge(Li) detectors will soon be available. In Saclay we are now using a 50-cm³ Ge(Li) detector, and therefore we can use smaller samples, such as separated isotopes. However, there are resolution problems with the larger detectors. In these experiments, an increase of the neutron flux is justified, but improvements in fabrication and resolution of solid state detectors can sometimes compete with the flux. I think it is clear that a neutron source at least 10 times more intense than the Saclay linac would allow one to perform a lot of experiments in the neutron energy range from 1 eV up to 1000 eV with 100 g of material. One has to keep in mind that to increase the thickness of the sample to get more capture events is not always a good solution, because it is then difficult to resolve neutron resonance doublets and also to

normalize the data (e.g., partial radiative widths).

For experiments with neutron energies above 1 keV, the number of capture events is weak since the main process is then neutron diffusion. For light nuclei ($A \approx 50$), the ratio Γ_n/Γ_γ is ≈ 100 to 1000, so that very long flight paths are needed to investigate up to 50 to 100 keV. This type of experiment will be very fruitful since resonances with different angular momenta ($l = 0, 1, 2$) can be studied. In this case, one has to have very good neutron resolution, and therefore one has to have very intense neutron sources.

To conclude, a linac about 20 to 50 times more intense would allow one to perform angular correlation experiments on the gamma-ray spectra of fissile nuclei, and to use polarized neutrons associated with a polarized target. But for this type of experiment, improvements in detector resolution and efficiency can probably compete with higher fluxes."

BARTHOLOMEW: I have made a similar estimate for ING to see how far in excitation energy one could go with a NaI-Ge(Li) coincidence arrangement. One has to assume a lot of numbers, of course--say, for example, a 50-cm³ Ge(Li) detector and a large sodium-iodide crystal, a source of 100 cm² that would be black to these neutrons, the usual solid angles, etc. One could then expect to do such coincidence experiments with the storage ring source to something like 10 keV. With the other ING source, I estimated the limit to be a little under 1 keV.

Now we shall take up the last item, concerning fast neutron reactions, and here none of the members of the Panel has had any direct experience. However, Bollinger has some remarks that he actually wanted to make following the last Panel.

BOLLINGER: It seems possible that by being too vigorous in my previous statement I introduced some misunderstanding in the minds of some of our foreign guests. Therefore, I want to give part of my speech over again, if you do not mind. There are just four points I want to make. First, in almost all of what I was saying, I was talking about the energy range from roughly 100 keV to 2 MeV. Second, when I stated that the best available white source, which I chose to be the Karlsruhe cyclotron, and the pulsed Van de Graaff were roughly equivalent, I meant this to be a rather rough statement. By roughly equivalent, I meant that for equal resolution the intensities ought to be equal within, say, a factor of 5. The reason I cannot narrow it down more than this is, as I explained, my argument is based entirely on my memory of two spectra, one of which I have not seen for some time. I went on to say that because of the ability of the electrostatic accelerator to do a much larger variety of experiments, it is the one I would select if I had to choose a single accelerator. On the other hand, this does not in any sense imply that the white source is useless; in fact, if it should turn out that the Karlsruhe machine has several times more intensity than the electrostatic accelerator, then for certain limited classes of experiments it would be

the machine to use. I hope this slightly less vigorous statement makes my position clear.

BARTHOLOMEW: One topic that has not really been covered is fast neutron reactions at still higher neutron energies. I would like to ask Hopkins of Los Alamos if he would say a few words on this subject.

HOPKINS: I want to make a few general comments on some of the things that were mentioned earlier this afternoon. First, I would like to bring up the subject of "figure of merit" again. The comment that I want to make involves the $1/\tau^2$ term of expression that has appeared in a number of papers and the one that Bollinger discussed earlier. In the region above a couple of MeV, we are limited in time resolution by beam pulse length, electronic resolution, detector time resolution, and several other variables. We try to maximize the resolution by varying these parameters. If we had twice the beam-burst length, then we would be down in our figure of merit by a factor of 4. We could, however, get the same resolution by moving back a factor of 2. On the other hand, if we had twice the beam-burst length, we could increase our detector thickness by a factor of 2, and increase our efficiency by a factor of 2 without losing time resolution. Then the final figure of merit would be reduced by only a factor of 2. So, instead of $1/\tau^2$, I suggest it goes as $1/\tau$, at least in this region for neutron detection above an MeV or so.

The second point that I want to make about the figure of merit is that one ordinarily assumes that there are roughly the same number of people working on each experiment. This, when applied to experiments such as those described by Diven and Hemmendinger, usually results in a very high figure of merit. However, one problem peculiar to the bomb experiments is that they have a very large number of people associated with these projects. This is a disadvantage, and I suggest that one should also put in a factor of $1/m$, where m is the number of individuals necessary to make an experiment "go."

In the MeV region, I agree with Harvey's comments that for the reactor cross-section work we do not need more intense sources. One major problem is that in the absolute cross-section measurements we are limited, by about a factor of 2 in the standard deviation, by the assay accuracy of foil materials. This comment also applies to the cross-section measurements of Diven and Hemmendinger. These problems have to be solved before we need to worry about neutron sources for these measurements. Another point I want to make concerns the white vs accelerator sources. I think that another example in favor of a white source is the bomb source. It has a very high intensity, of course, and good cross-section measurements have been made out to several hundred keV. These measurements can be carried on up to 14 MeV. Some experiments that are very difficult--for example, inelastic scattering--might conceivably be performed. However, I still think that it is a very difficult experiment to do. It seems to me that one of the biggest disadvantages to the bomb source is that it is not readily available. This certainly has to enter into anyone's calculation on

what kind of an accelerator to get, or what kind of a source to use. At the other end of the scale is the Mather plasma source, which sounds attractive because it is monoenergetic and very inexpensive. Perhaps "bi-energetic" would be a better term since the plasma source provides 14-MeV neutrons from the (d-t) reaction and 2.4-MeV neutrons from the (d-d) reaction.

No one has talked very much about doing neutron experiments up to 800 or 1000 MeV, which we can and will do with some of the large accelerators. I would put particular emphasis on such experiments as triple-scattering measurements, which are totally impossible now, multiparameter experiments, plus many of the other experiments which have been mentioned. A simple example is neutron-proton scattering, which is certainly flux-limited in the several hundred MeV region.

BARTHOLOMEW: Before throwing the meeting open to discussion, I would like to say a few more words about the "figure of merit." I think that we at Chalk River are probably the ones that have been associated with, or have perpetrated, this "figure of merit" that people use, and I would like to show a figure given at last years' Antwerp meeting that demonstrates that we really have not been unaware of the dangers of comparing all devices using the same figure of merit (Fig. 2). As Havens said, it is a good idea in comparing devices to include the peak intensity, which is indicated by I_p , and the average intensity, which is indicated by I_a . As you can see, we have been showing these parameters at the same time as the figure of merit. Also, there are three groupings on the diagram--the first one includes only the linac, the next one corresponds to fission-type devices, and the last group includes spallation-type devices. Thus, we also feel that the "figure of merit" is a very crude basis for comparison and should be used with caution, particularly in comparing between different classes of devices. Now the meeting is open to discussion.

VERVIER: I would like to give some further support to the last statement of Hopkins. In answer to a question last week at Gatlinburg, Bethe said that the most interesting experiments in the nucleon-nucleon interaction and, of course, in the neutron-nucleon interaction, were in the region of 50 MeV or, more broadly speaking, between 20 and 100 MeV.

BARTHOLOMEW: If there are no further comments, I think I should summarize some of the points that were brought out in this discussion. From the number of topics, one might get the impression that in the future the largest nuclear physics field will be with thermal neutrons. I think this is illusory, reflecting somewhat the interest of the Panel. Eventually, much more nuclear physics will no doubt be done with medium- and high-energy neutrons, as we learn to use these neutrons as precise tools in the same way that we now use charged particles. As has been said, we still have much to learn about nuclear forces, and most of this will be done at high energies, to which we have devoted only a very small fraction of our time. But of course there is still much

Parameters of Very Intense Neutron Sources

Facility	$\frac{n}{\text{MW sec}}$	τ sec	f sec ⁻¹	I_p sec ⁻¹	I_a sec ⁻¹	N	M sec ⁻³
R.P.I. Linac	3×10^{15}	9×10^{-9}	720	4×10^{17}	3×10^{12}	4×10^9	3×10^{28}
PHERMEX (44 MeV)*	8×10^{15}	2×10^{-7}	750	3×10^{19}	4×10^{15}	5×10^{12}	1×10^{29}
	"	6×10^{-9}	"	9×10^{19}	4×10^{14}	5×10^{11}	1×10^{31}
Harwell Linac (booster)	2×10^{16}	1×10^{-7}	400	1×10^{17}	5×10^{12}	1×10^{10}	3×10^{26}
G.A. Linac (pulsed fast booster)*	3×10^{16}	1×10^{-5}	100	5×10^{20}	5×10^{17}	5×10^{15}	5×10^{27}
I.B.R.	3×10^{16}	4×10^{-5}	83	6×10^{16}	2×10^{14}	2×10^{12}	1×10^{23}
SORA*	3×10^{16}	5×10^{-5}	100	2×10^{18}	8×10^{15}	8×10^{13}	3×10^{24}
NRU(chopper)*	3×10^{16}	5×10^{-6}	103	2×10^{15}	1×10^{13}	1×10^{10}	4×10^{23}
Harwell SC	7×10^{16}	1×10^{-8}	200	5×10^{18}	1×10^{13}	5×10^{10}	1×10^{29}
Karlsruhe SFC	$\sim 8 \times 10^{16}$	2×10^{-9}	2×10^4	7×10^{17}	2×10^{13}	1×10^9	1×10^{31}
Hypothetical SFC*	$\sim 8 \times 10^{16}$	2×10^{-9}	10^5	8×10^{18}	1×10^{15}	1×10^{10}	5×10^{32}
Nevis SC	1×10^{17}	2×10^{-8}	60	1×10^{19}	1×10^{13}	2×10^{11}	3×10^{28}
LAMPF*	2×10^{17}	1×10^{-8}	90	2×10^{19}	2×10^{13}	2×10^{11}	2×10^{29}
ING(microstructure)*	3×10^{17}	3×10^{-9}	10^4 (5×10^7)	6×10^{19}	2×10^{15}	2×10^{11}	2×10^{32} (1×10^{36})
ING(storage ring)*	"	1×10^{-7}	200	4×10^{23}	9×10^{18}	4×10^{16}	9×10^{32}

* = Projected capability

SC = synchrocyclotron

SFC = sector focused cyclotron

$$I_a = I_{prf} = Nf$$

$$M = Nf\tau - 2$$

Fig. 2

nuclear physics to be done at thermal energies, and the Panel, I think, concludes that we need continuous high-flux sources, coupled with great attention to improving the beam quality and to careful design of experiments. Improvements in design or layout of the reactors or other devices would of course include beam tubes of special types such as tangential tubes, through-holes, etc., designed to reduce the background.

There are certain applications for pulsed sources in the thermal range, but in general one wants the continuous source. In the resonance range, much nuclear physics is going to be done with resonance capture gamma rays. In general, one needs only enough resolution to separate resonances in this work. In the energy region less than 1 keV, accelerator pulsed fission devices may be best, although a steady-state reactor with a chopper may be good below, say, 50 or 100 V. At energies above several keV, an accelerator alone would be preferred. For fast neutron reaction work in the high energy range, one would of course be more interested in the spallation-type devices.

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